

Deterministic Casualty Analysis of the Pebble Bed Modular Reactor
for use with Risk-Based Safety Regulation

by
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B.S., Mechanical Engineering
University of New Mexico, 1993

Submitted to the Department of Ocean Engineering and the
Department of Nuclear Engineering in Partial Fulfillment for the Degrees of

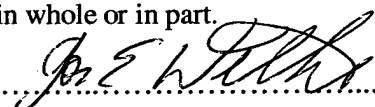
Master of Science in Naval Architecture and Marine Engineering
and
Master of Science in Nuclear Engineering

at the
Massachusetts Institute of Technology
September 2002


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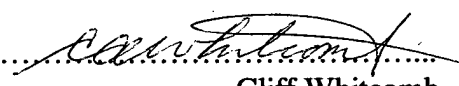

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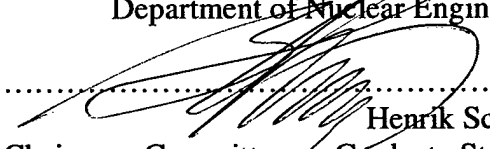
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DETERMINISTIC CASUALTY ANALYSIS OF THE PEBBLE			4.	
RED MODULAR REACTOR FOR USE WITH RISK-BASED			5.	ent
C. MONITOR REPORT NUMBER SAFETY REGULATION				
BY: JON E. WITHEE, MIT				
D. PREPARED UNDER CONTRACT NUMBER				
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Submitted to the Department of Ocean Engineering on August 9th, 2002
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Abstract

The resurgence of interest in the use of nuclear technology for electrical power production has resulted in a desire to improve the existing licensing structure. Improving the licensing structure will result in reduced design time and cost for new reactor plants. An improved regulatory process is also necessary in order to license advanced reactors that are not light water technology. Risk based reactor licensing, which uses the Probabilistic Risk Assessment (PRA) to justify most licensing questions, is a proposed replacement for the current methods.

This work further develops the risk-based regulatory process by analyzing a portion of a new reactor concept. A reactor similar to the Pebble Bed Modular Reactor (PBMR) is the design chosen for the analyses. The designers of the PBMR assert that the reactor's inherently safe design justifies the use of a non-standard containment system. This assertion can be treated as a design question to be justified using the risk-based approach. The effect of the changing the containment system is incorporated into the PRA for the PBMR.

The contributions to the event and fault trees of the PBMR are determined for two casualties that affect the plants decay heat removal system. The initiating event for both of these casualties is assumed to be a beyond design basis earthquake.

The first casualty is steam blanketing of the reactor vessel due to a rupture in the Reactor Cavity Cooling System (RCCS). This casualty is shown to have little effect on the safety of the plant. The second casualty was failure of the RCCS due to operator inaction. If this casualty were to occur the reactor vessel has the possibility of failing catastrophically. The failure of the reactor vessel could result in damage to the fuel and release of radionuclides. The probability of this casualty resulting in a significant release of radionuclides is $7.5 \cdot 10^{-11}$ / year. For the two casualties evaluated in this work, the use of a non-standard containment is justified.

Thesis Supervisor: Michael Golay
Title: Professor of Nuclear Engineering

Biographical Note

Jon Withee, a student in the Ocean and Nuclear Engineering Departments since June 2000, received his Master of Science in Naval Architecture and Marine Engineering and Master of Science in Nuclear Engineering in September of 2002.

Jon received a Bachelor's degree in Mechanical Engineering from the University of New Mexico in May 1993. He made the Dean's List. As an undergraduate, he was awarded the Bryce C. Rowan Award for Military Engineers. He then served the United States Navy as a submarine officer. Erek then went on to be a supervisor at Ballston Spa Nuclear Power Training Unit. He plans on obtaining a doctoral degree in Ocean Engineering at Massachusetts Institute of Technology.

Acknowledgments

I am indebted to my wife, Paun, and my children, Lauren and Garet, who have been very understanding and supportive during the last two years of my education. They have sacrificed in order to allow me to have the required time and space to concentrate on my work.

I would like to thank my thesis advisor, Professor George Michael Golay, and my thesis reader, Professor Cliff Whitcomb, for their support in this effort. Their counsel and guidance were vital.

I appreciate the United States Navy for giving me the opportunity to develop my academic and professional background by allowing me to attend Massachusetts Institute of Technology. I also want to thank Fred Silady of Technology Insights and Stan Ritterbush of Westinghouse, for their assistance in my research.

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Chapter 1 Goal and Outline for this Work

This chapter states the goal and the problem statement for this work. Then the motivation behind the goal and the plan for this work are discussed. An outline of this work is also provided for understanding the scope of the project.

1.1 Goal for this Work

The Nuclear Regulatory Commission (NRC) is interested in improving the means by which reactor plants are designed and licensed. Risk-Based goals and requirements have been given serious attention due the advances made in Probabilistic Risk Assessment (PRA) techniques. The NRC has sponsored a project to determine if it is possible to use a risk based approach for licensing nuclear reactors. This Risk-Based approach uses the PRA as the primary measure of reactor safety. Developing a new means of licensing reactors is particularly important when considering reactor types that are not based on light water reactor technology. Any improved licensing method must be able to be applied to advanced reactor types in order for it to be of use by both the Nuclear Regulatory Commission and the nuclear reactor designers.

Analyzing the safety of a reactor design is a large task. This is an even greater problem when the reactor is different from those for which there is past experience. There are three goals for this work. The first goal is to identify casualties specific to the Pebble Bed Modular Reactor. The second goal is to perform the deterministic analysis on these casualties to determine any possible effects on reactor safety. Finally, probabilities are assigned to any outcomes for use in a Risk-Based analysis of the Pebble Bed Modular Reactor.

1.2 Problem Statement

An improved licensing method must allow for the development and construction of new reactor types in the United States. Risk-Based Safety Regulation is a possible method by which advances in reactor technology can be incorporated into the United States nuclear industry. In an effort to prove the viability of the integrated probabilistic approach this

work attempts to use Risk-Based methods to analyze a gas cooled reactor. Eskom, a South African utility company, is designing an advanced gas cooled reactor called the Pebble Bed Modular Reactor (PBMR). This reactor is designed to be inherently safe. The designers of PBMR state that a standard containment system is unnecessary to ensure the protection of the public due to the reactor's safe design. This work will explore the validity of this claim using the risk-based approach. The response of the reactor will be analyzed for two casualties that affect its decay heat removal capability.

1.3 Motivation and Plan for this work

The United States nuclear industry has been bogged down since the incident at Three Mile Island. An adversarial relationship has arisen between the Nuclear Regulatory Commission and the nuclear industry. Designers are sometimes hard pressed to justify some portions of their reactor designs to the NRC. Reactor designers and NRC officials sometimes cannot agree on the requirements. These disagreements result in longer certification times and addition of unnecessary redundancy in some instances, both of which lead to increasing the cost of the reactor plant. This research is being conducted on behalf of the NRC.

The improved licensing structure will use the Probabilistic Risk Assessment (PRA) as the primary means of certification. Currently, the PRA is used to justify design decisions. In the improved licensing structure the NRC will specify PRA standards for various events. Then the engineers will have specific safety goals to meet while designing the reactor plant. When the PRA is completed, all uncertainty in the analysis will be addressed with additional testing, defense in depth, and increased safety factors. When the time comes to have the NRC review the design, the engineers will only have to justify their PRA calculations and their handling of the uncertainty. The approval process should thus be made less adversarial.

A Light Water Reactor (LWR) has previously been analyzed using the integrated probabilistic approach. There is a large amount of data available relating to LWR. The analysis proved that an integrated probabilistic method could be a useful tool. But in order for the method to be of real benefit it must be able to be applied to advanced reactor types.

There is very little experience and data associated with advanced reactors in the United States. A new licensing structure must be able to be applied to these designs. In this work, the contributions to the event and fault trees for a portion of a new reactor concept are determined. A reactor similar to the PBMR is the design evaluated in this work.

The PBMR will be analyzed using the risk-based analysis method. First, the overall design is evaluated for its response to a large earthquake. Then the possible casualties that could result in fission product release to the environment are discussed. Next, two casualties are chosen for in-depth analysis. Then, the effect of this casualty on the reactor containment is evaluated. Finally, probabilities and uncertainties will be assigned to the various aspects of the casualty in order for them to be incorporated into the PRA of the PBMR to help determine if a standardized containment system is necessary.

1.4 Report Outline

Chapter 2 describes the procedure of reactor licensing using an integrated probabilistic approach.

Chapter 3 is a description of the Pebble Bed Modular Reactor. This chapter includes a description of the systems and specifications. The relevant subsystems are also discussed.

Chapter 4 is an analysis of the possible casualties that could lead to fission product release and a description of the problem that is used in this case study.

Chapter 5 describes the methods by which the various analyses are performed. The finite element analysis program used to analyze the heat transfer of the PBMR is HEATING. This chapter explains how HEATING works. Then, the analysis model of the PBMR is described. Finally, the deterministic methods used for the heat transfer and structural response are described.

Chapter 6 presents the results of the analysis. The results detail the effects the casualties have on the vessel temperature, core temperature, and structural response.

Chapter 7 is a discussion of how the deterministic results are related to the probability of fission product release.

Chapter 8 presents the conclusions of this work and discusses further work necessary to complete the integrated probabilistic analysis of the PBMR.

Chapter 2 Description of the Integrated Probabilistic Approach to Reactor Licensing

This chapter discusses the problems with the current licensing structure. Then the goals for nuclear regulatory reform are stated. Finally, the method by which probabilistic analysis can be used to improve the licensing of nuclear reactors is discussed. The majority of the information in this chapter is based on a presentation given at the June 2001 ACRS Workshop titled "A New Risk-Informed Design and Regulatory Process"[1].

2.1 Motivations for Regulatory Reform

One of the primary motivations for reforming the regulatory system is that the enormous capital costs and financial uncertainties associated with licensing and constructing nuclear power plants have restricted U.S. utilities from ordering any Nuclear Power Plants (NPP) in the past two decades. In the deregulated electricity market, nuclear power plants are even less economically attractive because these start-up costs hinder the possibility of a timely return on investment. On the other hand, some socio-economic factors have simultaneously increased the appeal of nuclear power as a viable option for power production. Compared to the fossil fuels used by coal, gas turbine, and natural gas plants, nuclear fuels cost utilities far less per megawatt. This disparity in cost grows each day as fossil fuels become scarcer and more expensive. Additionally, nuclear power provides electricity without the immediate environmental impacts of fossil fuel-burning plants. The U.S. government has repeatedly voiced its concern over America's growing dependence on foreign oil. Increasing electricity consumption, stricter air pollution standards and continuing unrest in the oil-rich Middle East has forced the U.S. to diversify its energy production profile [2]. The Department of Energy and power production industry have indicated that nuclear power will continue to play a vital role in the U.S. energy profile. In order to spur new NPP construction, the start-up costs for a NPP must be reduced, while still maintaining adequate public protection.

Public opposition to nuclear power has by no means vanished but rather waned in the absence of high-profile safety incidents since Three-Mile Island. Environmental,

political, and economic turmoil surrounding fossil fuels has further drawn the spotlight of criticism away from nuclear power. Additionally, the commercial nuclear power industry boasts an excellent and well-documented safety record. As a result, the nuclear power industry argues that it often endures unjustified regulatory burden without significant or demonstrable safety benefit. Accordingly, the Nuclear Regulatory Commission has accelerated efforts to re-evaluate and reform those regulations governing the licensing, construction, operation, and decommissioning of commercial nuclear power plants. The NRC hopes to reduce the enormous capital costs preventing utilities from building NPPs, while still adequately protecting the public from the unlikely, yet possible, dangers of nuclear power. Furthermore, the existing regulatory framework cannot accommodate some advanced light-water reactors (LWR) and non-LWR types currently being considered as new projects by utilities within the U.S., pending results from foreign projects. Thus, the U.S. urgently needs a new regulatory approach to licensing nuclear power plants. This improved regulatory framework must continue to ensure adequate public protection while reducing costs and providing guidance for the regulation of advanced reactor types.

2.2 Problems with the Current Regulatory Procedures

The design and regulatory process that was employed traditionally and is still used today consisted of the following steps. First, the designer develops a plant design that both produces power reliably and operates safely. The designer uses high level regulatory criteria and policies as inputs. Next, the regulator reviews the design. The designer and regulator then engage in a dialog. This dialog involves specifying safety features, their performance criteria, and methods of design and analysis. Throughout this process the designer documents the design basis while the regulator documents the safety evaluation, policies established, and criteria for future reviews.

Any reform of the regulatory process needs to help overcome the problems associated with the current structure. These problems are:

- The certification process is extremely long.
- Regulation is primarily concerned with deterministic calculations without explicit consideration for uncertainty.

- Regulatory decisions on specific issues are made in isolation, resulting in an inefficient and incoherent licensing structure.
- Deterministic analysis combined with subjective judgment has led to the practice of layering on of redundant.

This layering on of redundant systems is called defense in depth. This practice has been one of the primary means used to address regulators concerns. In some cases it is warranted, but in other cases unnecessarily drives up the construction and operating cost of plants. The goal of this work is to determine if it is possible to use a risk-informed design process to evaluate a portion of an advanced reactor design.

2.3 Goals for Regulatory Reform

The current regulatory policies have resulted in designers having difficulty getting a reactor certified when the design differs from those that have been previously certified. The result of these difficulties is a nuclear industry that has been unwilling to invest in advances in reactor design, even when design improvements could result in much safer operations. Light Water Reactors (LWR) have a great deal of technological inertia in the United States nuclear industry, due to the fact that all current requirements, test procedures, and manufacturing infrastructure are based on LWR. This technological inertia combined with inefficient regulatory structure has resulted in very slow progress in our nuclear industry. The Goals for regulatory reform are:

- Create methods to assure consistency of nuclear power plant applicants and regulators in performance and goals for producing safe, economical power.
- Change the adversarial nature of the current licensing structure.
- Create a licensing structure that is adaptable for a wide range of reactor types.

2.4 Description of Probabilistic Regulatory Approach

In recent decades, the techniques of PRA have reached a level of maturity and acceptability justifying the extensive incorporation of PRA in any new regulatory philosophy. In essence, PRA calculates the frequency (events per year) with which

undesirable sequences of events will occur. Certain known combinations of low-level events (pipe breaks, valve failures, operator errors, etc) must occur in order to cause a given high-level event (core damage, radionuclide release, etc). The frequencies of high-level events can then be calculated using fault tree or event tree logic and the frequencies of low-level events. The frequencies of low-level events from historical data, traditional deterministic analyses, and expert opinion are calculated. Hence, PRA provides us with a manner for quantifying safety, rather than relying upon vague terminology such as 'not safe', 'safe enough' or 'extremely safe' [3].

The overall goal for safety and regulatory reform for the nuclear power industry should be to create methods to assure consistency of nuclear power applicants and regulators in performance/goals for producing safe, economical power plants. Successful electric power production is based on two key issues. First, the production of electrical power must be economically feasible; otherwise it will be replaced by another means of production. Second, the power must be produced in a safe manner. Power production that poses a significant risk to the public or environment is not acceptable. Both of these issues must be accounted for in any design and regulatory procedures.

The theory behind a risk informed design and regulatory process is that all regulatory decisions should be based on the informed beliefs of decision makers. These beliefs can and should be stated in a probabilistic format. The basic equation governing a probabilistic determination of an event with a given Probability Density Function (PDF) is:

$$\text{Probability } (x < X < x + dx) = f(x) dx \text{ (for a continuous distribution)}$$

The regulator acceptance criteria must reflect acceptable best-estimate performance expectations and uncertainties. Regulatory questions and acceptance criteria should also be stated within a probabilistic framework. This framework should be as comprehensive as possible [1]:

- Utilize probabilistic and deterministic models and data where feasible. Use subjective judgment where these models are not feasible.
- State all subjective judgments probabilistically and incorporate them into the PRA.

- Require both license applicant and regulatory staff to justify their decisions explicitly.
- Initiate resolution process to resolve applicant-regulator disagreements.

The risk informed regulatory approach for nuclear power plant evaluation is performed probabilistically and is supported by deterministic analyses, tests, experience, and judgments at all conceptual stages of development. Safety results of defense-in-depth, performance margins, best-estimate performance, and subjective judgments are all incorporated into a comprehensive PRA. The PRA is used to state the evaluator's beliefs concerning system performance. The level of detail of acceptance criteria becomes finer as the level of concept development increases. Figure 2.1 is a diagram of the Risk-Informed Design and Regulatory process.

Current licensing structures require reactor designs to be proven safe for Design Basis Accidents (DBA). Unfortunately, these DBAs are based on Light Water Reactors (LWR) designs. When a reactor design differs greatly from the existing LWRs, some of the design basis accidents become irrelevant. Also, different reactors designs will have to be analyzed for casualties that do not apply for LWR. For example, a High-Temperature Gas Reactor (HTGR) is susceptible to a water ingress casualty which is meaningless for a LWR. A risk-informed licensing structure must be adaptable for various designs. Reference [1] provides a description of how to develop casualties for new reactor designs.

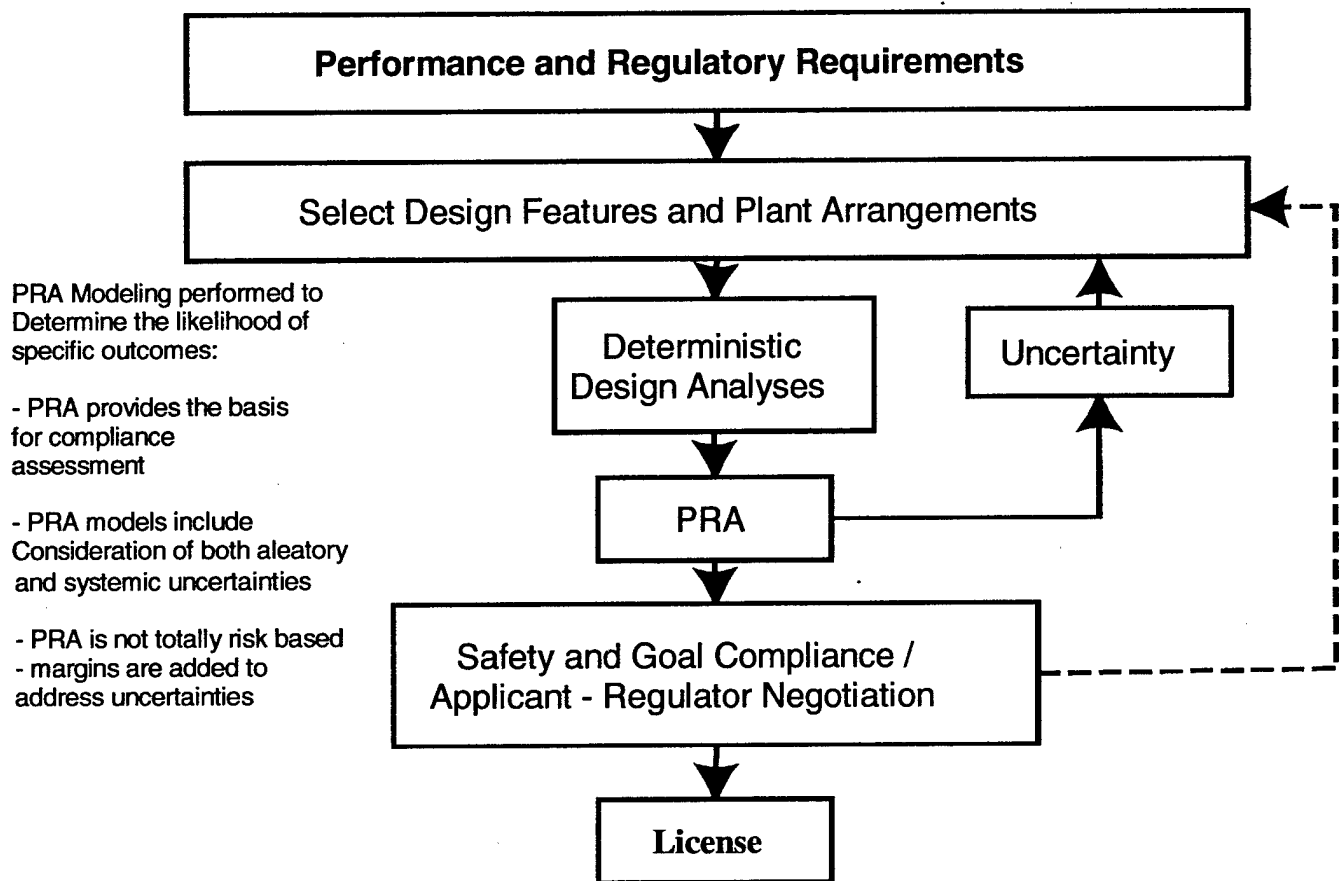


Figure 2.1 Risk-Informed Design and Regulatory Process [1]

Chapter 3 Description of the Pebble Bed Modular Reactor

The analysis of this work is performed using a reactor similar; but not identical to the PBMR. The major characteristics are the same for both the notional reactor and the actual PBMR. This chapter contains a description of the PBMR. First, a brief history of the pebble bed technology and how it came to used is given. Then the current reactor design is discussed. Finally, the support systems relevant to this work are described. The information provided in this chapter is extensive and additional information can be found in references [4, 5].

3.1 History of the Pebble Bed Reactor Type

Eskom Ltd., the nation utility of South Africa currently has about 34,000Mwe of generating capacity, primarily coal plants with some hydro power and two nuclear plants located in southwest South Africa near Cape Town. There is a growing demand for electrical power in the region. Eskom compared several alternatives for power generation and decided that LWR were too expensive. It was decided that coal fired power plants were not advantageous since the power demand is growing in the south and all of the coal is located in the north of the country.

Eskom decided to use German pebble bed technology, which was an ultra-safe system, so they could concentrate on the economics of the system. Modular plants of a modest size were selected to take advantage of economical benefit of many identical systems. As power demand increases additional power units can be added. Eskom and their partners envision that the PBMR technology could be exported world-wide because of its safety and attractive economics. The partners in this venture British Nuclear Fuels (BNFL), Excelon, and Industrial Development Corporation of South Africa (IDC) [4, 5]. For further information on the PBMR please see the Eskom website.

3.2 General Description of the Pebble Bed Modular Reactor

The PBMR concept is based on German high temperature helium cooled pebble bed reactor technology demonstrated at the AVR and THTR reactors. The PBMR is a modular, graphite moderated, helium cooled, pebble bed type reactor that uses a Brayton direct gas cycle to convert the heat into electrical energy by means of a helium turbo-generator. A regenerative heat exchanger, called a recuperator, is used to improve the thermodynamic efficiency. The PBMR plant specifications developed by Eskom are given below in Table 3.1 [5].

Table 3.1 PBMR Plant Specifications

Maximum Power Output	268 MW thermal
Continuous Stable Power Range	0-100%
Anticipated Cost (nth module)	\$1000 / KWe
Construction Schedule	24 Months
Outage Rate	2% planned & 3% forced
O&M and Fuel Costs	\$4-5 / MWhr
Emergency Planning Zone	< 400 meters
Plant Operating Life Time	40 years
Maximum Operating Temperature	900 ° C
Maximum Operating Pressure	7 MPa

The PBMR has a large core with a low power density. The helium coolant remains in a gaseous phase and is inert. The layout of the PBMR reactor plant is shown in Figure 3.1. The plant is a traditional closed gas cycle turbine system using a recuperator to improve the thermodynamic efficiency. The Power Control System varies the power of the reactor by changing the amount of helium in the system.

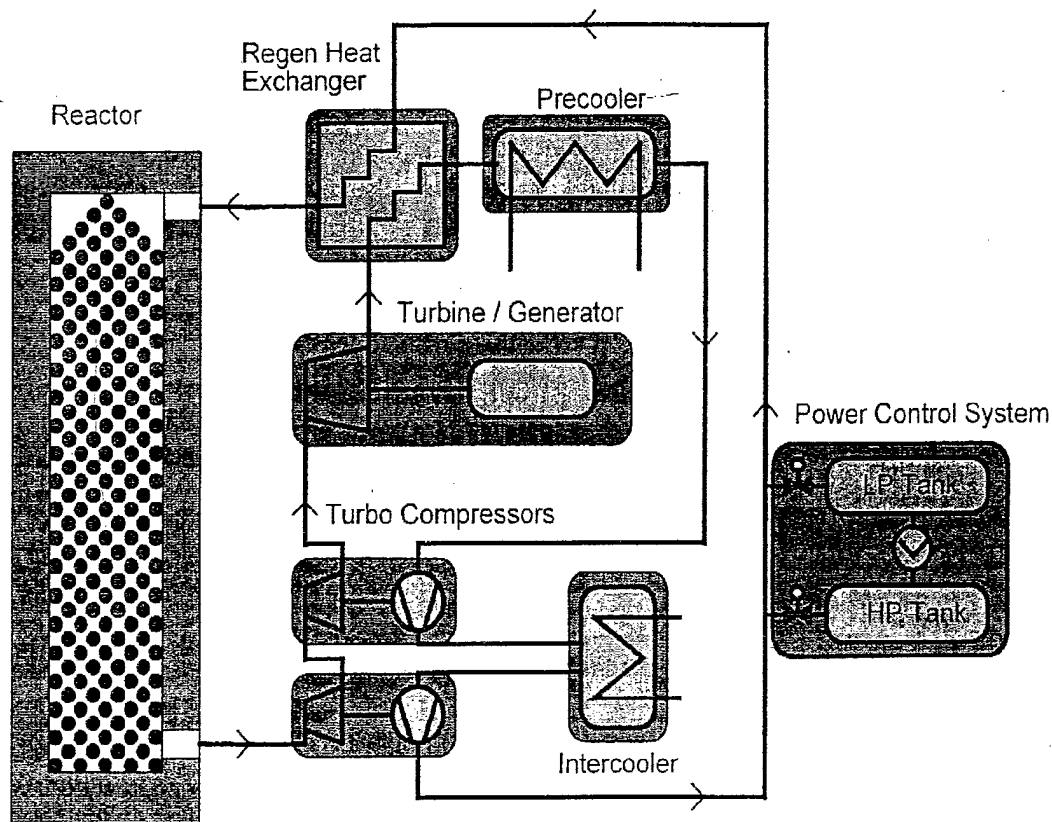


Figure 3.1 PBMR Plant Layout [5]

3.3 Description of the Pebble Bed Reactor Fuel Design

One of the most important safety features of the PBMR is the fuel design. The PBMR uses spherical pebbles 60 mm in diameter, each impregnated with about 15,000 enriched (8% U-235) uranium particles coated using the three-layer TRISO particles shown in Figure 3.2. These fuel particles can withstand very high temperatures prior to releasing fission products. The German test facility indicated that the fuel particles can withstand temperatures of about 1700 °C with no significant damage and up to 2200 °C with damage limited to about 0.01% of the fuel. Figure 3.3 shows the results of the German tests of the TRISO fuel particles [5].

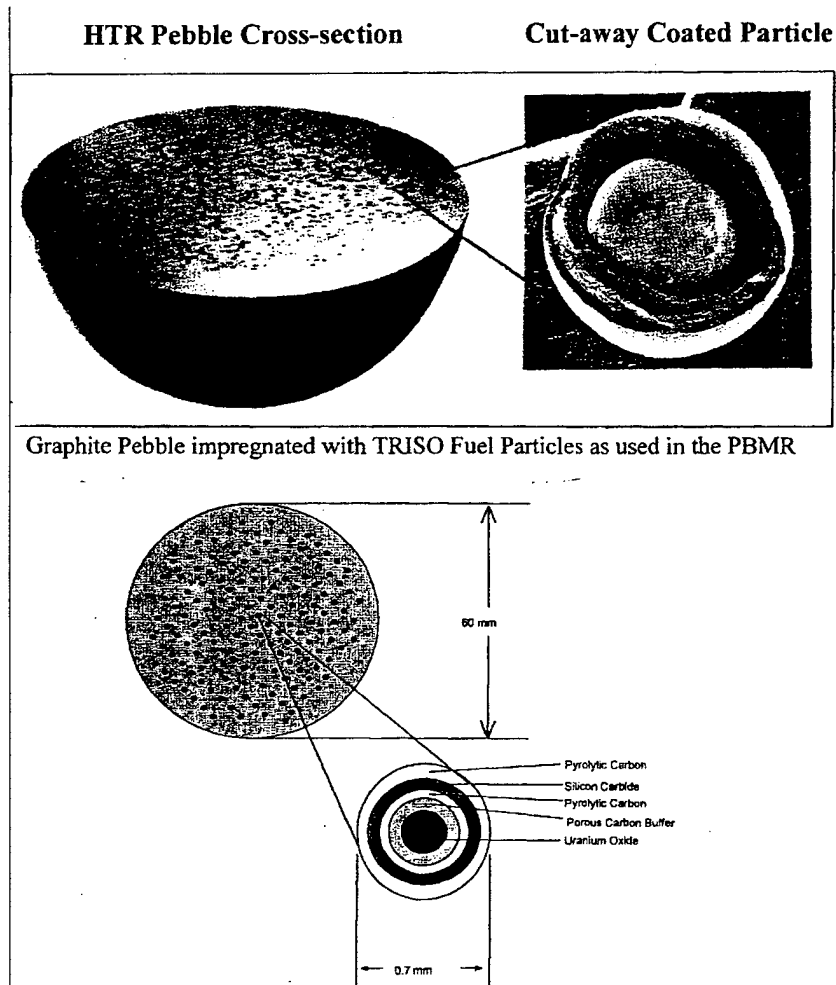


Figure 3.2 PBMR Graphite Pebble and TRISO Fuel Particle [5]

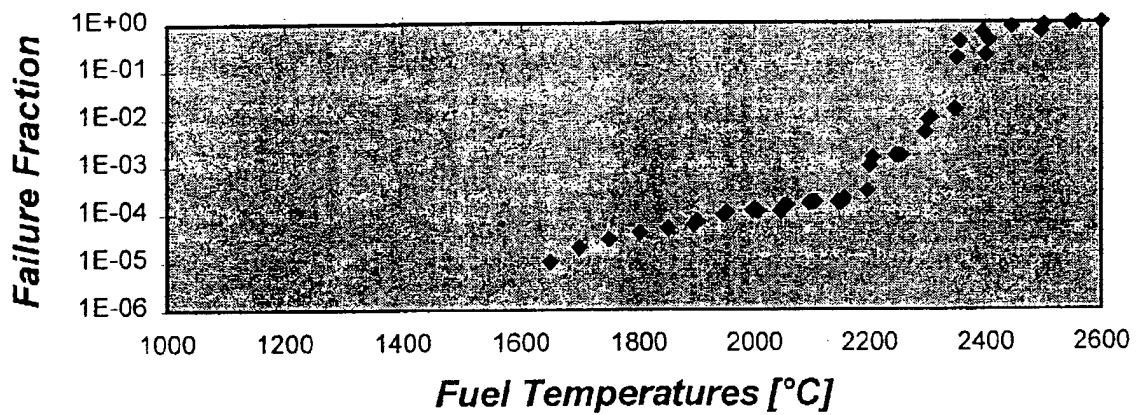


Figure 3.3 Results of TRISO Fuel Particle Tests [5]

3.4 PBMR Core Description

The PBMR core is 3.5 meters in diameter and 8.5 meters high. The core consists of two regions. The cylindrical inner region is filled with approximately 110,000 moderating balls which are only graphite. The fuel balls are housed in an annular region that surrounds the center reflector. There are approximately 336,000 pebbles in the fuel region. This arrangement flattens out the power distribution. About 5,000 pebbles are cycled through the core each day. They are monitored for fuel burnout; in order to determine if they should be sent back to the top of the core or to the spent fuel holding area. The core has a large negative temperature coefficient, which adds to its safety.

The core is surrounded by a graphite reflector. The reactivity control and shutdown system consists of two subsystems. The primary means of reactivity control are control rods. These rods are fully inserted to shutdown the reactor. At power, the control rods primary purpose is to make up for xenon transients. The second subsystem is the Small Absorber Spheres (SAS), which are used as an emergency backup system.

3.5 Overall Plant Layout and Containment System

The plant will consist of a single building approximately 50 x 26 meters and 42 meters in height, with 21 meters below ground level. The reactor will be contained in a citadel. The citadel will house the reactor inside the Reactor Cavity. The turbo-compressors and Power Turbine will be contained within Power Conversion Unit (PCU). Figure 3.4 is a side view of the PBMR building. This figure shows the positions of the Reactor Cavity and the PCU. The containment system is also shown in Figure 3.4.

In the event of a large rupture in the reactor cavity, the containment system prevents over-pressurization by relieving pressure into the adjacent PCU. If pressure is still too high, lift plates in the PCU relieve pressure by venting citadel into the building. The building itself has pressure relief dampers that relieve excess pressure to the atmosphere. The sequential barriers posed by the containment system act to contain fission products first to citadel and then to the building. There is also a pressure relief shaft that has a rupture disk which actuates to protect the citadel from small to medium

size ruptures; it is equipped with a filtration system. Eskom analysis shows that this design in combination with the inherent safety of the PBMR design prevents the need for a standard containment system [5]. According to Eskom, the requirement of a standard containment system would make the PBMR economically unfeasible.

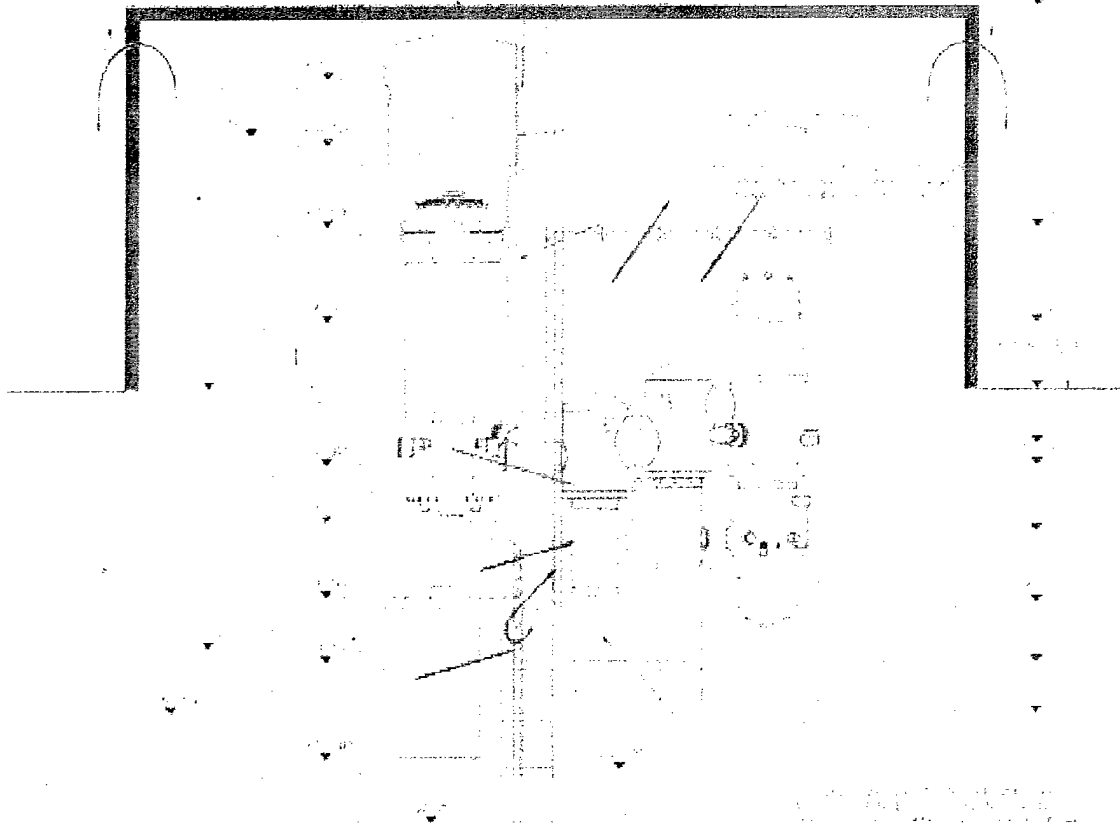


Figure 3.4 Plant Layout and Containment System [5]

3.6 Description of the Reactor Cavity

The analysis portion of this work deals primarily with the casualty response of the PBMR. Proper modeling of the reactor cavity is very important for accurate response calculations. The reactor cavity contains several systems. For this casualty analysis the most important components to be modeled are the reactor and the Reactor Cavity Cooling

System (RCCS). The reactor vessel is supported by mounts that allow for thermal expansion and contraction. The reactor is unlagged in belt region. This allows for heat to be removed from the reactor vessel via radiation and convection. In normal operation water circulates through the RCCS removing heat from the reactor cavity. The reactor and its attached subsystems are sometimes referred to as the Reactor Unit.

Figure 3.5 is a cutaway view of the reactor cavity showing the location of the various components. This figure is based on data obtained from the PBMR Safety Analysis Report (SAR) [5]. The RCCS consists of 45 cylinders that are 22.5 meters tall. The system consists of three identical and independent subsystems. The RCCS is one the primary design features that makes the PBMR inherently safe. A more detailed description of the RCCS is included in Appendix A [5].

3.7 Cooling Systems

Several cooling systems are required for proper operation of the PBMR. The ultimate heat sink for the initial PBMR test facility will be sea water. The sea water is distributed to the various other cooling systems via the Open Circuit Cooling System. Figure 3.6 shows the loads supplied by the Open Circuit Cooling System [5]. The backup cooling tower is used in emergency situations, where power is lost to the open circuit cooling pumps.

The RCCS normally cooled by the open circuit cooling water. The backup cooling tower supplies flow to the RCCS when electrical power is lost to the open circuit cooling pumps. The pumps that circulate water to the backup cooling tower are powered from emergency diesel generators. In the event of a loss of all electrical power, the RCCS system is designed to boil off over time removing heat from the reactor cavity. The PBMR SAR states that there is enough water in the RCCS to keep the reactor cavity cool for three days without flow of circulating water. The RCCS can also be supplied by fire hoses, if electrical power is unavailable. The ability of RCCS to remove heat from the reactor cavity, without any external motive force, is one of the design characteristics that make the PBMR and inherently safe design.

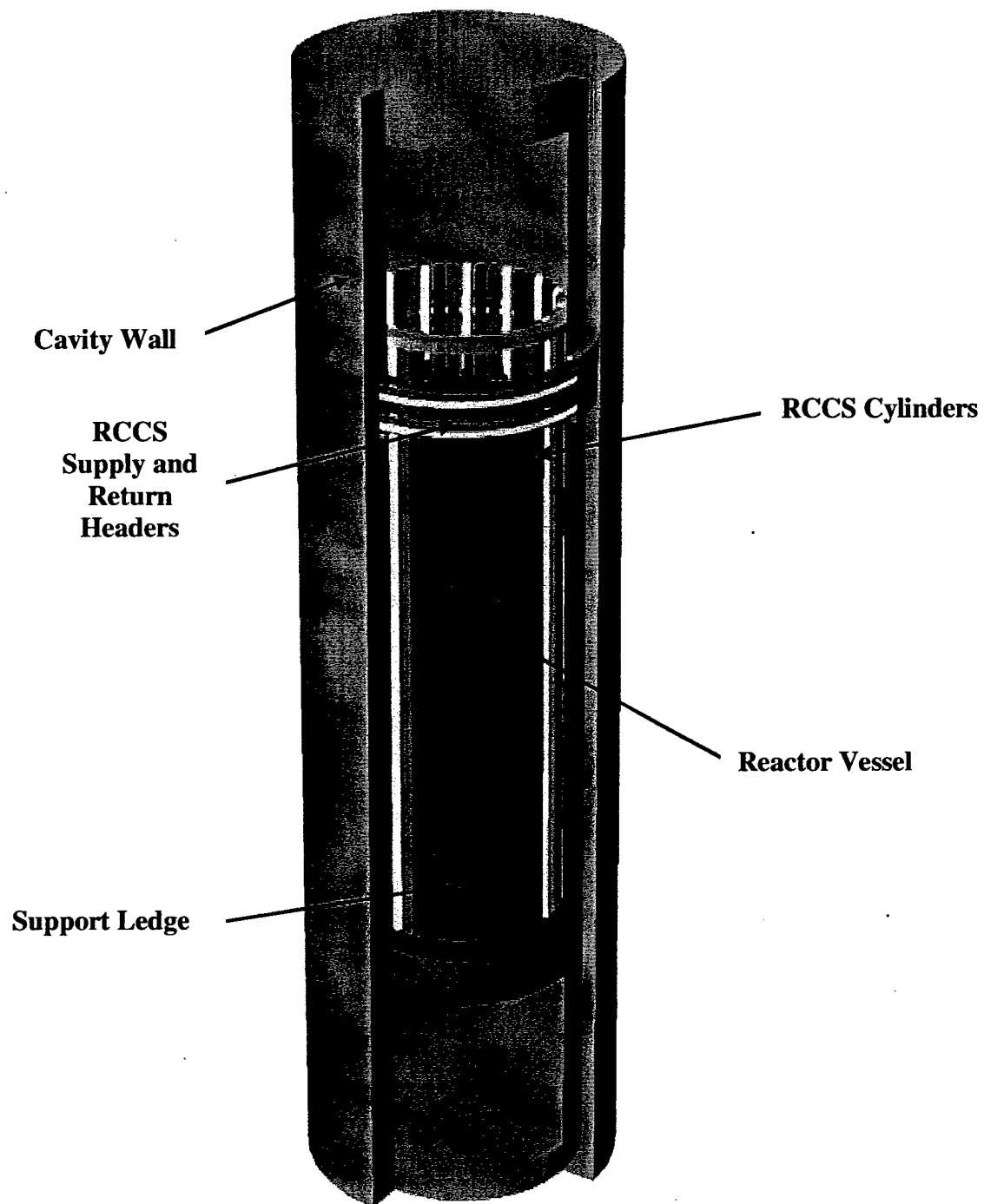


Figure 3.5 Side View of Reactor Cavity

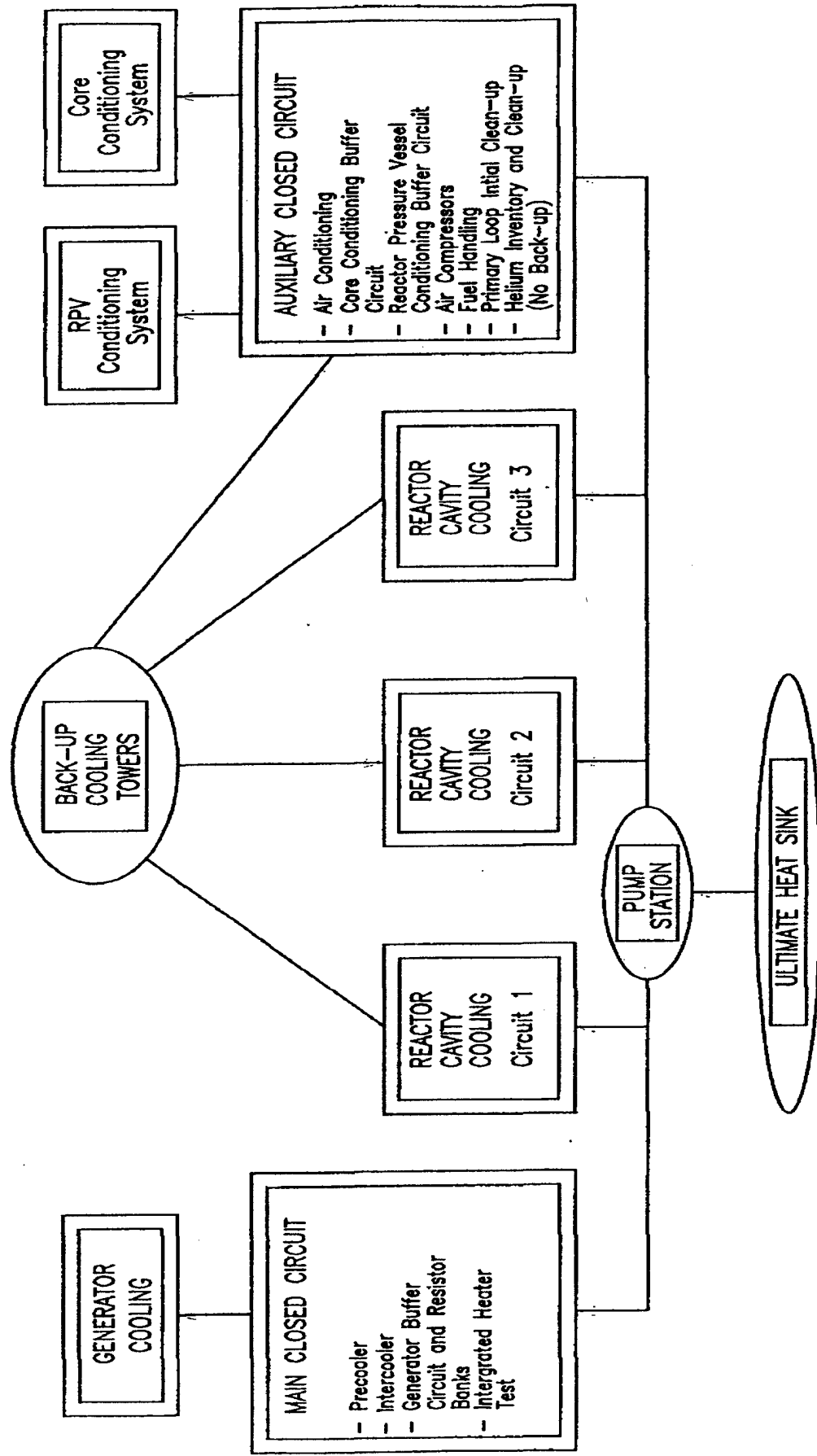


Figure 3.6 Active Cooling System Flow Diagram [5]

Chapter 4 Preliminary Casualty Analysis

The goal of any reactor designer is to have a final plant that produces safe and economical power. In order for a plant to be considered safe it must not pose a health risk to either the operators or the public. The primary concern for public safety is the release of radioactive contaminants from the reactor building to the surrounding environment. This chapter discusses the possible causes of fission product release from the PBMR and identifies the specific casualties for which deterministic analysis is performed.

4.1 Causes for Fission Product Release

All of the paths through which fission products could be released require multiple elements in the reactor design to fail. The possible means of fission product release are best described by a diagram. Figure 4.1 is a Master Logic Diagram for a notional nuclear reactor. As can be seen in the diagram the causes for fission product release are:

- Coolant Inventory Excursion
- Reactivity Excursion
- Temperature Excursion
- Pressure Excursion

Figure 4.1 is applicable for a light water reactor [1]. The casualties that can cause fission product release must cause damage to the fuel, escape of fission products from the primary pressure boundary, and escape of the fission products from the containment building. The same holds true for the PBMR. However, due to the design of the PBMR reactor design, damage to the fuel from high temperatures is unlikely. The high temperature characteristics of the fuel (see Figure 3.3) result in the vessel reaching its melting point long before the fuel exceeds a temperature where fission product release is likely.

Master Logic Diagram

Performance Goal Level

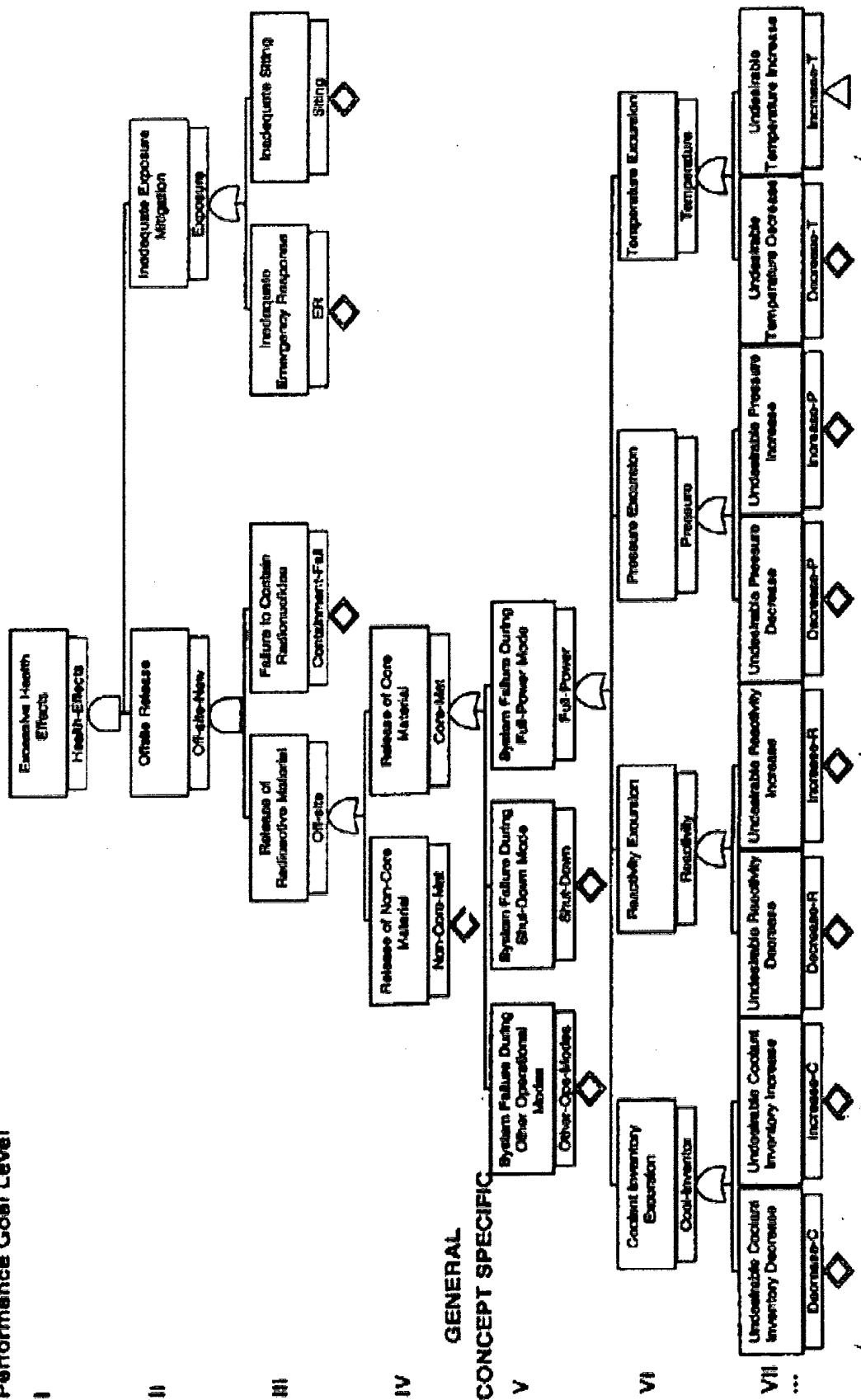


Figure 4.1 Master Logic Diagram [1]

For this research the initiating event of the casualty is assumed to be a beyond design basis earthquake. The possible consequences for this event are evaluated. Some of the possible casualties during an earthquake are: loss of all electrical power (normal and backup), incapacitation of operators, rupture of helium coolant system, collapse of building, and rupture of cooling systems. Casualties are chosen for analysis to determine if they will result in fission product release. Analysis by Eskom has shown that a Depressurized Loss of Coolant (DPLOC) casualty would not result in fuel damage [5]. The total collapse of the building is deemed unlikely. A reactivity excursion would not result from an earthquake due to the reactivity control and shutdown system. The fuel could be damaged by chemical attack if water or air entered the core, but neither of these casualties is likely due to the plant design.

The PBMR relies on the proper operation of the RCCS in order to remove decay heat that is generated in the core following the reactor scram. Impairing the operation of the RCCS is a possible cause of mechanical failure of the reactor vessel. Normally, the PBMR transfers its heat to the RCCS via radiation and convection heat transfer. Thermal analysis of the reactor shows that the fuel will not be damaged prior to vessel failure due to the high temperature characteristics of the fuel. However, catastrophic failure of the reactor vessel could result in fuel damage due to mechanical shock, thermal shock, or chemical attack. The deterministic portion of this research deals with two possible methods by which the passive decay heat removal of the PBMR could be impaired.

4.2 Description of the Chosen Casualties

For the first casualty the following is a sequence of events that are assumed to occur for which the deterministic analyses are applicable. First a loss of all electrical power occurs. Then reactor scrams. Operators at the plant are incapacitated. A rupture of one of the RCCS tubes occurs in the reactor cavity. This rupture sprays water onto the reactor vessel which is initially at 279 °C. This water flashes to steam. Steam fills the reactor cavity. The steam displaces air in the reactor cavity. Radiation heat transfer from the reactor vessel to the RCCS cylinders is reduced by the steam in the gap between the

vessel and RCCS. This reduction in heat transfer results in higher vessel temperatures than under normal operations.

The first deterministic analysis deals with determining the effect water vapor has on the overall heat transfer out of the reactor vessel. This is a casualty that has not been analyzed for by Eskom. Figure 4.2 is a flowchart of how the decay heat is transferred through the reactor and to the surrounding RCCS cylinders. The effect of having a damaged RCCS subsystem is shown. In Chapter 6 the effect on vessel temperature of the steam is shown to be negligible.

The second casualty chosen for analysis also involved the RCCS. Without pumps running the RCCS system operates by transferring decay heat to water in the cylinders. The water boils off and is vented from the reactor cavity area. The cylinder temperatures stay at approximately 100 °C, while the water is boiling off. The water in the cylinders can remove a great deal of heat from the reactor vessel without exceeding design temperature limits. The PBMR SAR states that the RCCS system should operate in the boil off mode for 3 days before it ceases to function due to loss of water. If the water is not replenished by operators the system ceases to function and vessel temperatures can start to rise. The second casualty determines the effect of operators failing to refill the RCCS.

The PBMR reactor vessel is made of SA 508 mild steel. Yield strength and modulus of elasticity of carbon steel vary with temperature. At elevated temperatures (> 600 °C) the yield strength is greatly reduced. Rupture of the vessel and subsequent damage to the fuel is possible if vessel temperature becomes too high. Fission products can then be released from the fuel and reactor. The rupture causes the pressure in the reactor cavity to rise and the pressure relief systems associated with the citadel complex will be actuated. The initial relief of the helium prior to fuel damage removes the primary motive force that could transport the radionuclides which are formed from damaged fuel.

Rupture in a RCCS Cooling Chamber during an Earthquake

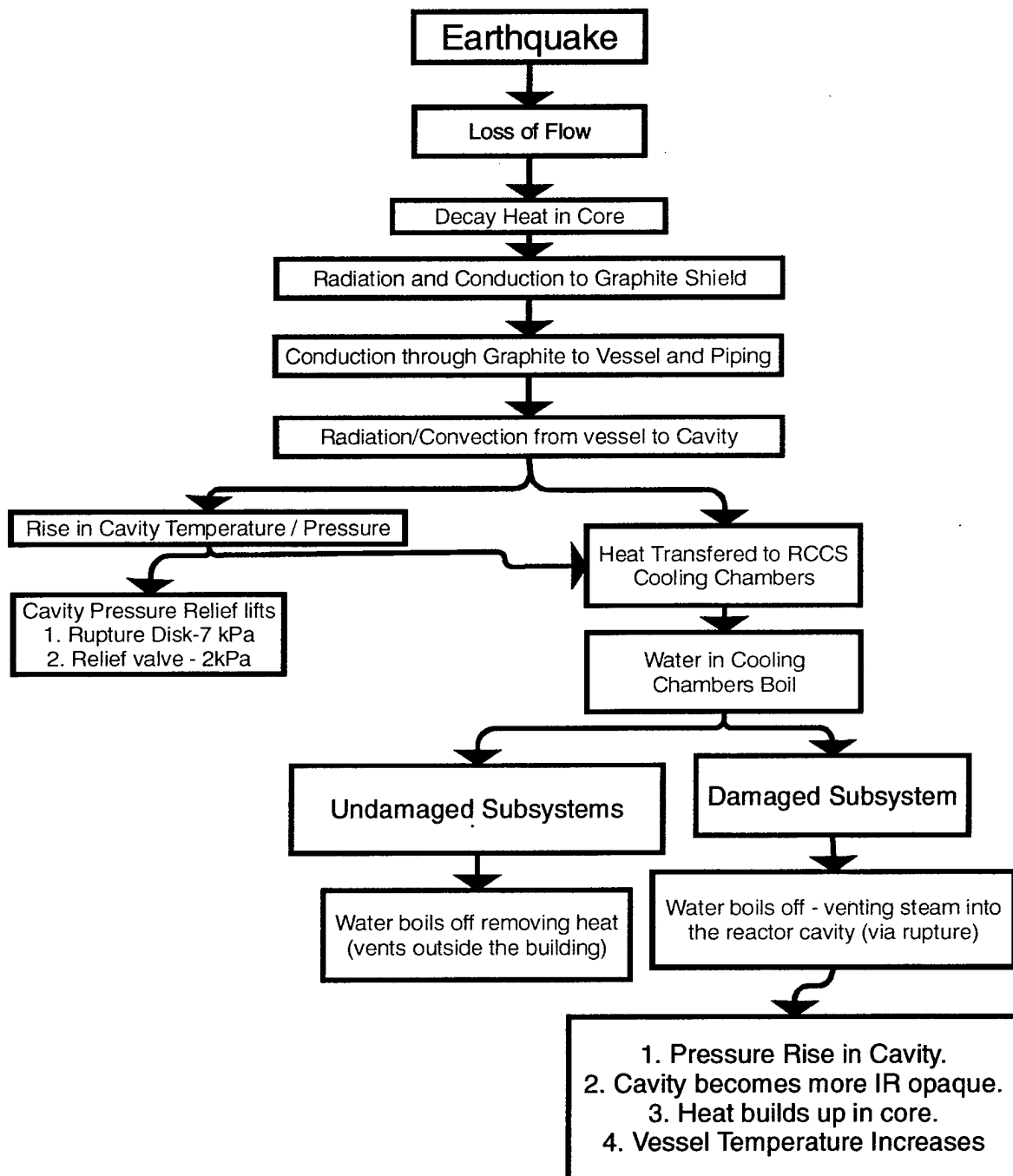


Figure 4.2 Damaged RCCS Casualty Flowchart

Chapter 5 Deterministic Methods Used

This chapter describes the methods used to analyze the casualty response of the notional PBMR. First the HEATING finite element analysis program is described. Next the heat transfer model used in the analysis is described. The analytical equations used in the heat transfer model are explained. Finally, a description of the reactor vessel structural equations is given. The notional PBMR is referred to as simply the PBMR in this and the following chapters.

5.1 Description of HEATING

The heat transfer analysis program used in this research was developed by Oak Ridge National Labs. A detailed description of the program can be found in reference [7]. "HEATING is a general-purpose conduction heat transfer program written in Fortran 77. HEATING can solve steady-state and/or transient heat conduction problems in one-, two-, or three-dimensional Cartesian, cylindrical or spherical coordinates. A model may include multiple materials, and the thermal conductivity, density, and specific heat of each material may be both time- and temperature-dependent. The thermal conductivity may also be anisotropic. Materials may undergo change of phase. Thermal properties of materials may be input or may be extracted from a material properties library. Heat-generation rates may be dependent on time, temperature, and position and boundary temperatures may be time and position dependent. The boundary conditions, which may be surface-to-environment or surface-to-surface, may be specified temperatures or any combination of prescribed heat flux, forced convection, natural convection, and radiation. The boundary condition parameters may be time- and/or temperature- dependent. General graybody radiation problems may be modeled with user-defined factors for radiant exchange. The mesh spacing may be variable along each axis. HEATING uses a run- time memory allocation scheme to avoid having to recompile to match memory requirements for each specific problem. HEATING utilizes free-form input." [7].

This work uses the finite element program to model the PBMR reactor cavity and its surroundings with cylindrical and cylindrical shell regions. HEATING has advantages

and disadvantages for these analyses. The advantage is that most of the components being modeled are actually cylindrical in shape. Unfortunately, HEATING can not account for the movement of fluid or conservation of mass. This limitation is a problem when the RCCS cylinders, which are filled with boiling water, are being analyzed. It is not possible to accurately simulate the boiling and mixing of water in the cylinders and the loss of mass due to boil off of the water. In one portion of this analysis, this limitation is overcome by using the total heat transfer between the reactor and RCCS to estimate the time at which sufficient water is boiled off to render the RCCS inoperable.

5.2 PBMR Heat Transfer Model

The reactor cavity is assumed to be surrounded by soil. This assumption was made in order to ease calculations. Because of the very slow heat transfer rate, the lower boundary of this model is assumed to be 26.63 m. There are 26 regions in the model. The total number of nodes in the model is 2342. Figures 5.1a and 5.1b are diagrams of the different regions of the heat transfer model.

Initial equilibrium cycle core conditions obtained from the VSOP code were assumed to exist at the time of shutdown. Table 5.1 shows the initial conditions used for the model. Where the temperature is spatially dependent, the figures that display them are identified in the table. The core modeled was the Eskom pebble-bed reactor being proposed in South Africa, which is being used by MIT as the reference core design. The KFA decay heat curve was assumed.

Table 5.1 PBMR Model Initial Conditions

Region	Initial Temperature (°C)
The Core	Figure 5.2
Top Reflector	Figure 5.3
Side Reflector	Figure 5.4
Helium Gap	279
Pressure Vessel	279
Air Gap	150
RCCS	100
Concrete Wall	50
Earth	35

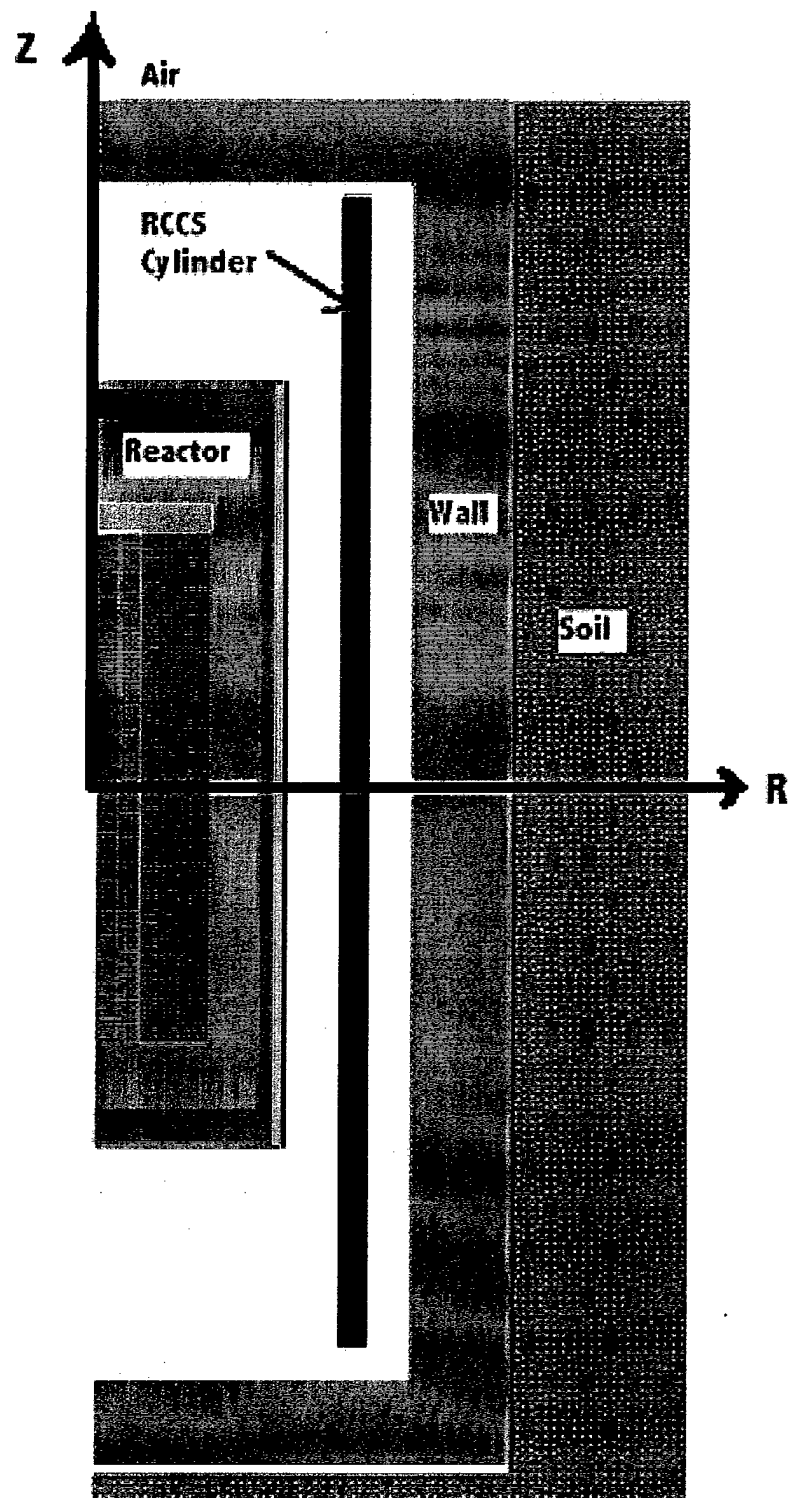


Figure 5.1a PBMR Heat Transfer Model Regions

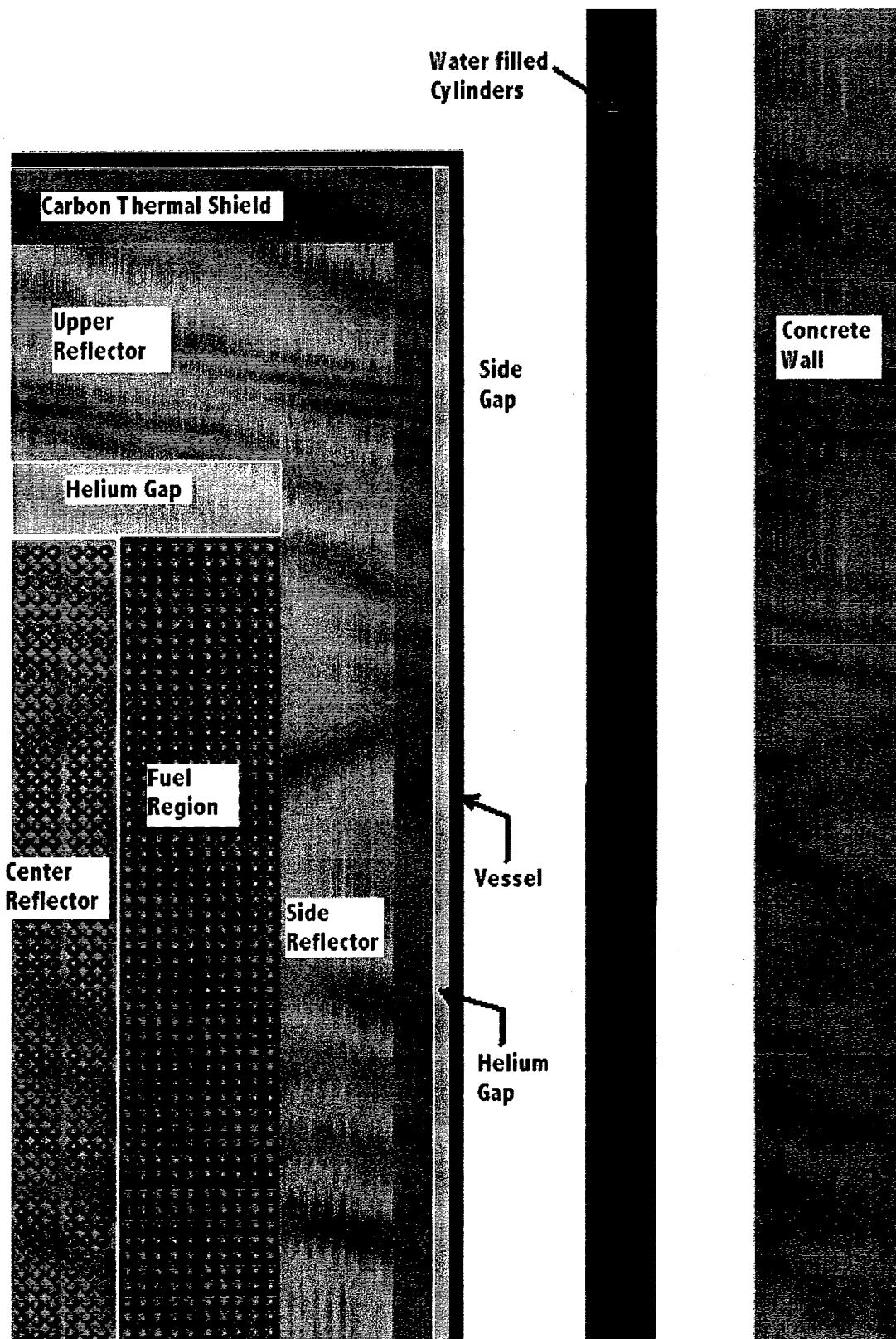


Figure 5.1b Detailed View of Heat Transfer Regions

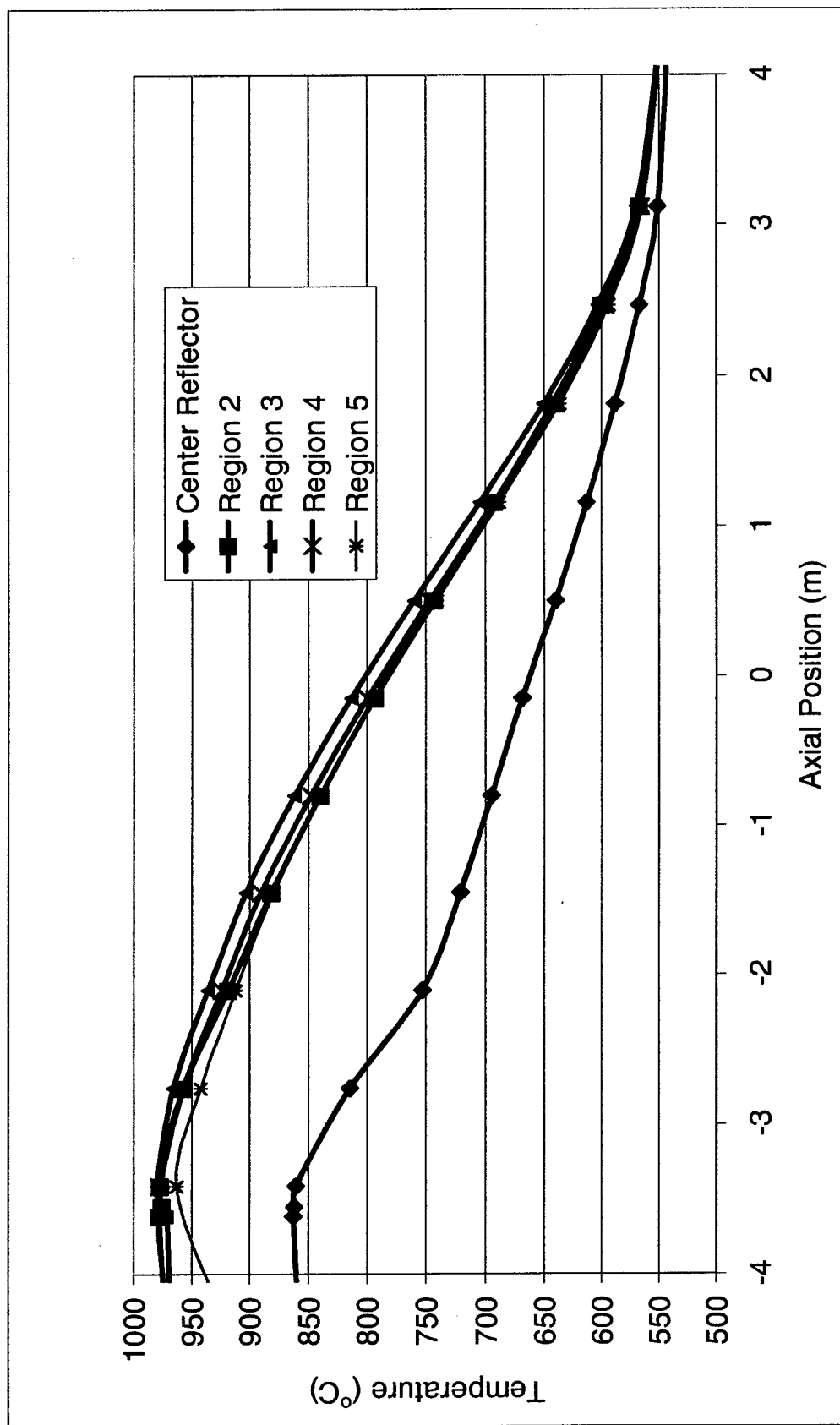


Figure 5.2 Initial Temperature of Core Regions

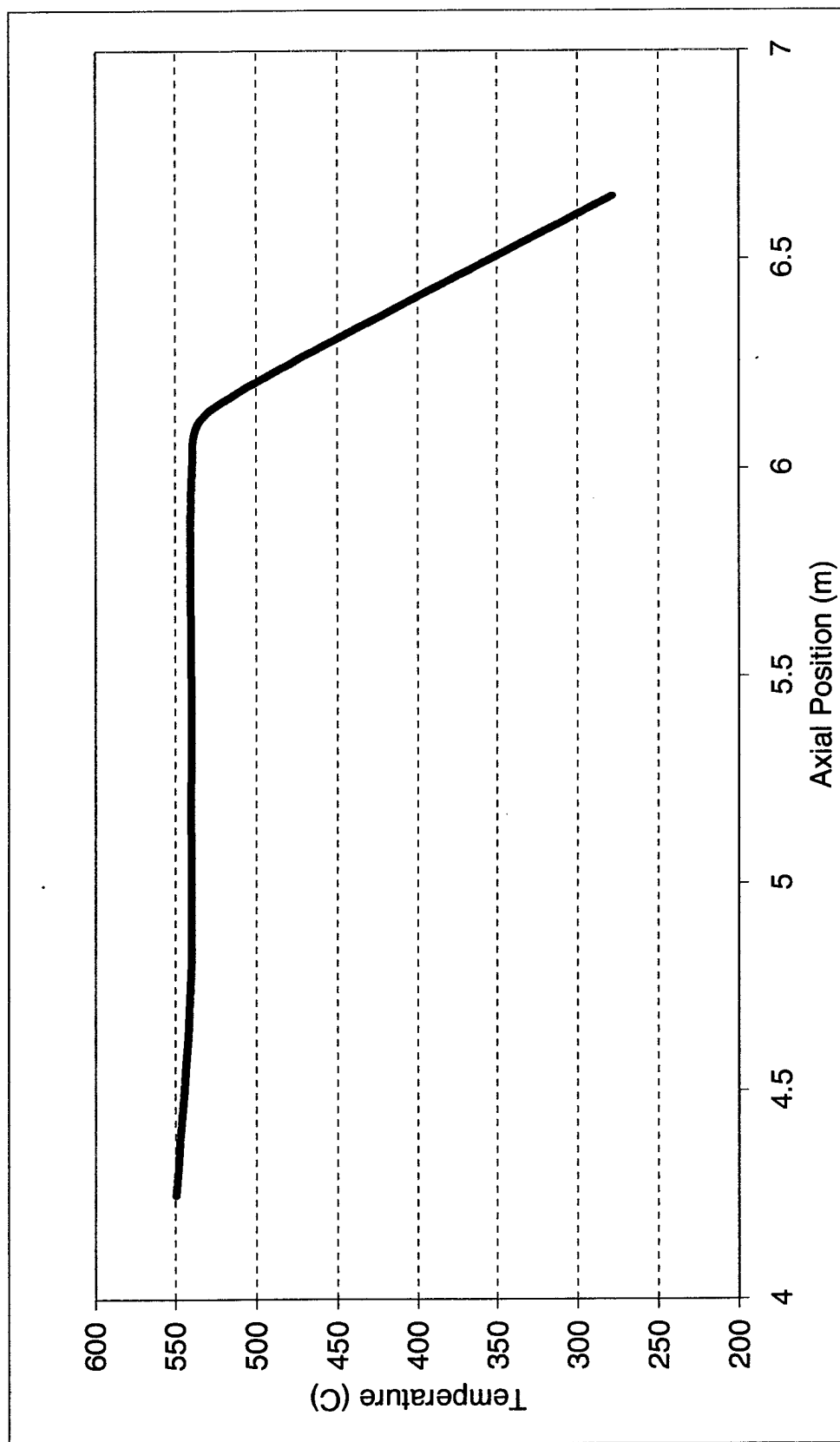


Figure 5.3 Initial Temperature of the Upper Helium Gap and Top Reflector

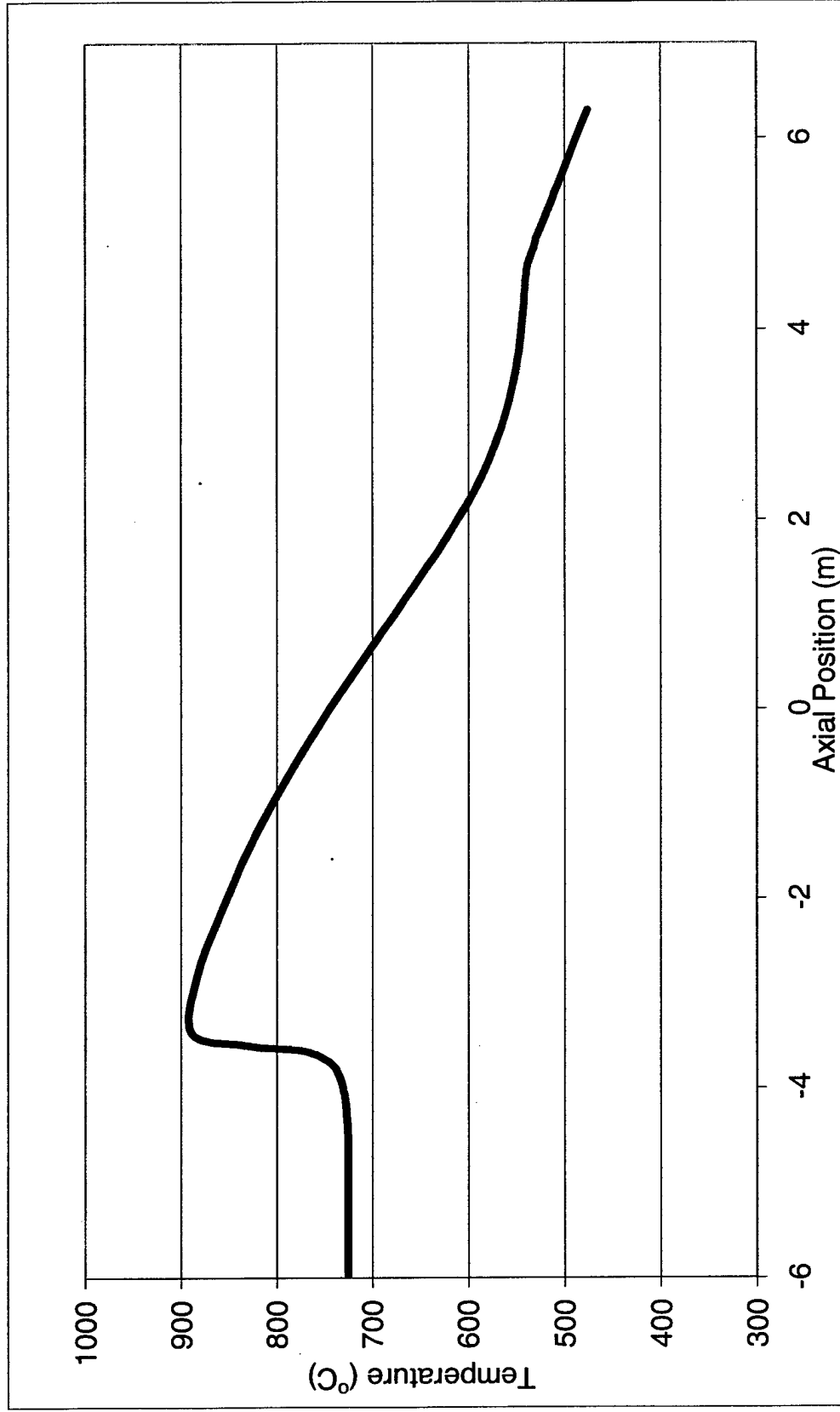


Figure 5.4 Initial Temperature of the Side Reflector

The core barrel region, which is made of steel of small thickness and high thermal conductivity relative to the other materials, is neglected because of its low thermal resistance. For the steam blanketing casualty, several data runs are conducted for various conditions in the gap between the reactor vessel and the boundaries. The goal for this portion of the analysis is to determine the effect water vapor in the reactor cavity has on the maximum fuel and vessel temperatures.

Decay Heat Generation

After reactor shutdown, fission power induced by delayed neutrons subsides rapidly and thereafter, the heat released by radioactive decay of fission products dominates the reactor power. The decay heat depends primarily on the operating history of the reactor, including the reactor power level prior to shutdown, and on the duration of the shutdown period. The following empirical formula from is used to approximate the power released by radioactive decay [8]:

$$Q_{DH}(t_0, t_s) = Q_T A (t_s^{-a} - (t_0 + t_s)^{-a}) \quad (5.1)$$

Where:

Q_T = reactor power prior to shutdown,

t_s = reactor shutdown time,

t_0 = reactor operating time, and

A and a are constants given for different time intervals in Table 4.3.

Table 5.2 Constants A and a in Eq. (5.1)

Time Intervals (seconds)	A	a
$10^{-1} < t_s < 10^1$	0.0603	0.0639
$10^1 < t_s < 1.5 \cdot 10^2$	0.0766	0.181
$1.5 \cdot 10^2 < t_s < 4.0 \cdot 10^6$	0.130	0.283
$4.0 \cdot 10^6 < t_s < 2.0 \cdot 10^8$	0.266	0.335

Figure 5.5 displays the decay heat curve with respect to time after shutdown. The power curves for the 5 core regions are shown in Figure 5.6.

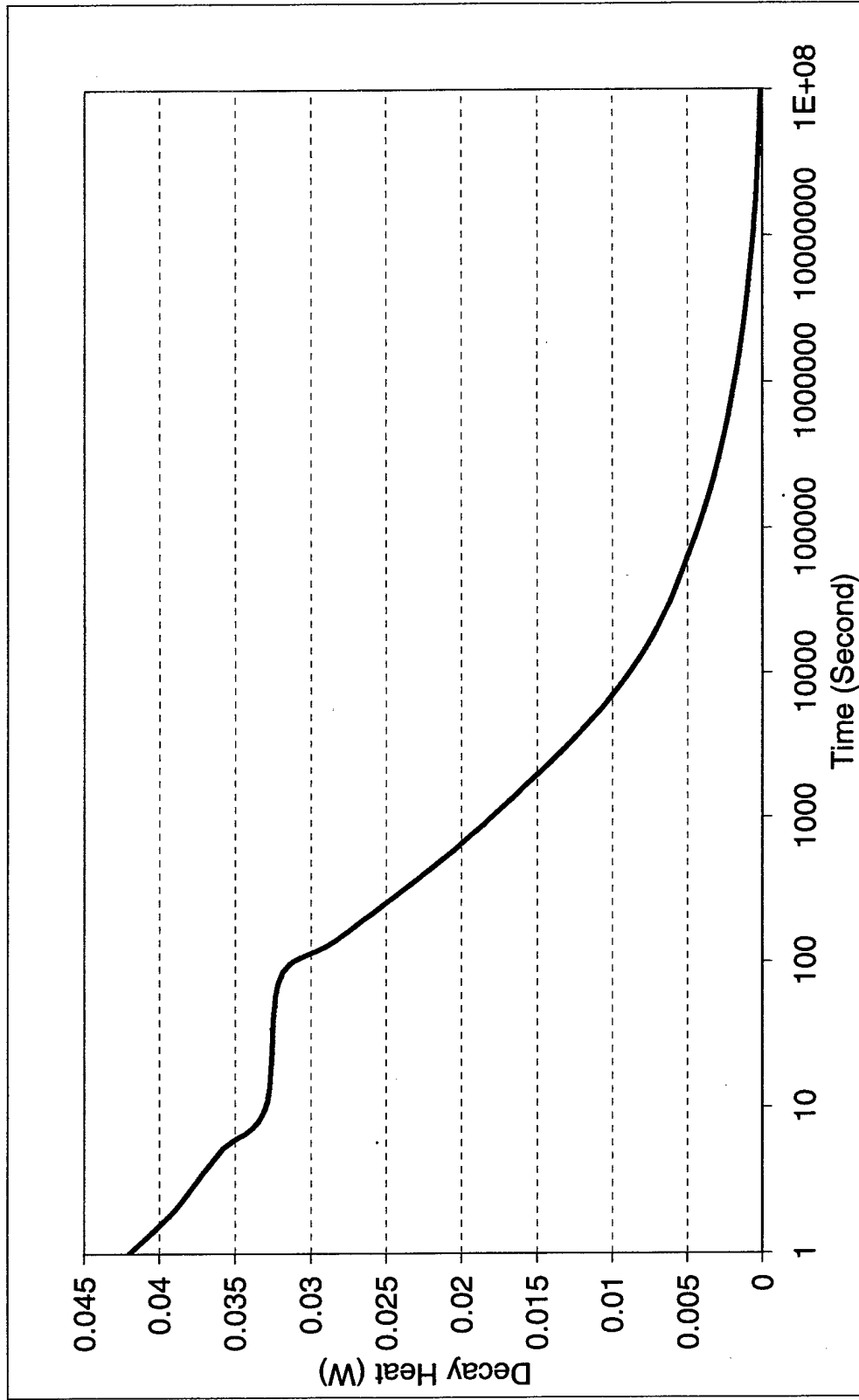


Figure 5.5 Decay Heat vs. Time

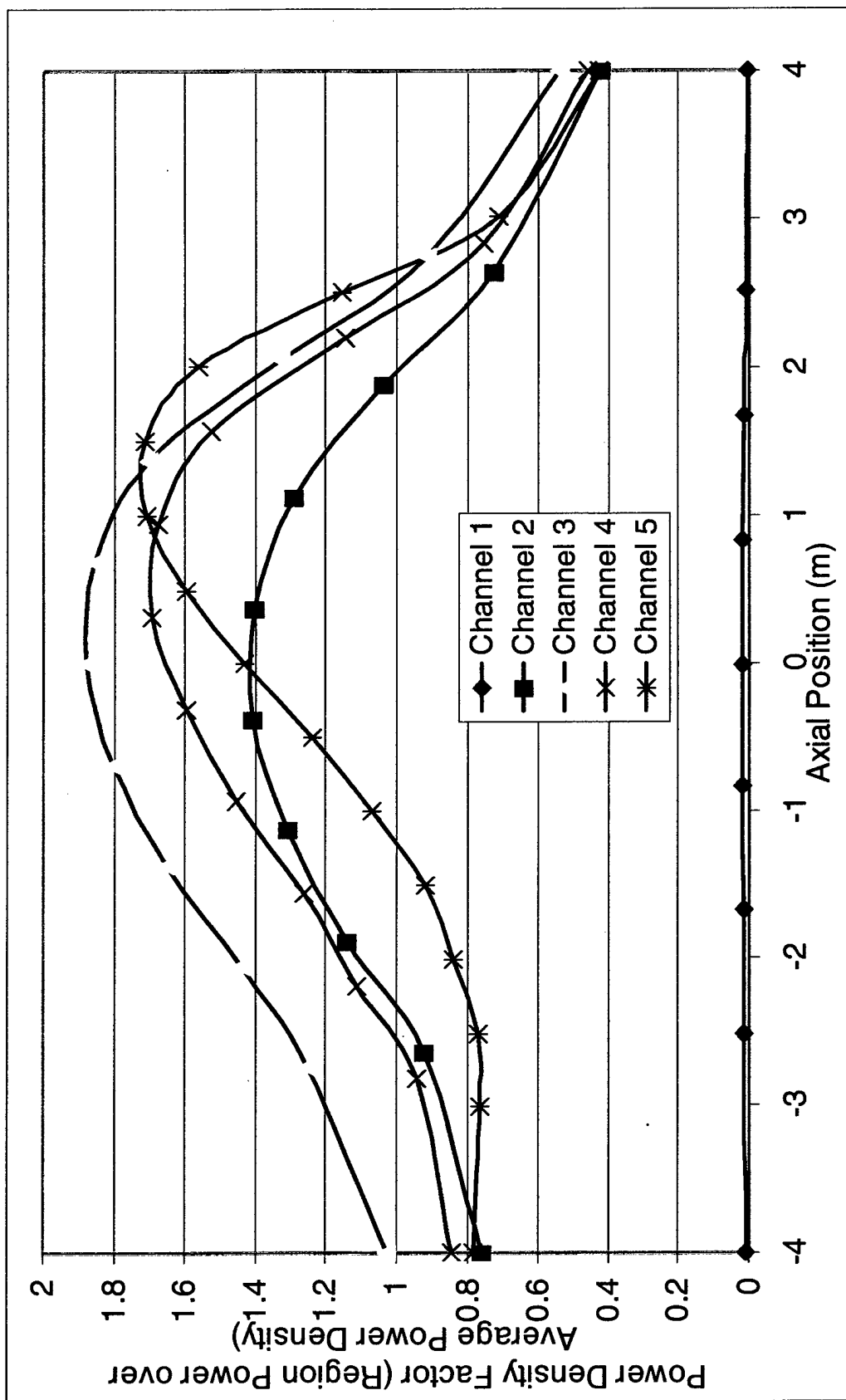


Figure 5.6 Power Densities of the Core Regions

The material properties used in this analysis are obtained from various sources [9, 10, 11, 12]. The properties required for input into the finite element analysis program are: specific heat, density, and conductivity. The values for some of these properties vary with temperature. Tabular values are entered into the input file so that the various material regions can be modeled properly. HEATING uses linear interpolation to determine values of properties between tabular input points. For the heat transfer calculations the RCCS was assumed to be a constant 100 °C for one of the casualties.

5.3 Radiation Heat Transfer

The modeling of radiation heat transfer through the gap between the reactor vessel and the RCCS cylinders was the most difficult portion of this analysis. Several methods were tried before a final analytical technique was chosen. Radiation heat transfer is the primary means by which heat is removed from the core. Convection heat transfer was found to be significant at normal operating temperatures but less significant at higher vessel temperatures. Conduction through the gap was found to be very small. Models in which only conductivity was used to cool the core showed that vessel and fuel temperatures exceeded design limits. This was due to the low densities in the gap between the reactor vessel and the RCCS cylinders. The total heat transfer through the gap was found by combining the calculated heat fluxes from radiation, convection, and conduction.

The radiation heat transfer through the gap was analyzed for various mixtures of steam and air. Air was effectively transparent to thermal radiation for this size of gap (.9 meters) between the RCCS cylinder and the reactor vessel. The amount of radiation transferred between two gray surfaces without the interference of an absorbing gas in between them is found using the following [11, 12]:

$$q_{net} = \frac{E_{vessel} - E_{RCCS}}{\frac{R_{vessel}}{R_{RCCS}} \times \left(\frac{1}{\epsilon_{RCCS}} - \frac{1}{2} \right) + \left(\frac{1}{\epsilon_{vessel}} - \frac{1}{2} \right)} \quad (5.2)$$

Where:

q_{net} : Heat Flux (W/m²)

E_{vessel} : Black Body Radiation of Vessel (W/m²)

E_{RCCS} : Black Body Radiation of RCCS (W/m^2)

ϵ_{vessel} : Vessel Surface Emissivity

ϵ_{RCCS} : RCCS Surface Emissivity

Blackbody radiation is found using the following equation

$$E_{Blackbody} = \sigma \times T^4 \quad (5.3)$$

Where:

σ : Stephan-Boltzmann Constant ($5.67 \times 10^{-8} W / m^2 K^4$)

T: Surface Temperature (K)

Steam is not transparent to thermal radiation. Therefore the heat transferred from the vessel to the RCCS cylinders varies with the amount of steam in the gap. The radiation heat transfer is affected by changes in the emissivity of the steam in the gap. Figure 5.7 is a graph of the total emissivity versus temperature for various pressure-thicknesses of steam [11]. The pressure-thickness is a measure of the amount of steam radiation must pass through. For this analysis the maximum pressure of water vapor was assumed to be 1 atmosphere, i.e. the reactor cavity is completely filled with steam. The pressure relief system for the citadel actuates at pressures slightly above atmospheric. The actuation of this system combined with the fact that other gases will always be present in the reactor cavity results in a maximum pressure thickness of .9 atmosphere-meters of water vapor, since the approximately .9 metres.

The steam attenuates some of the thermal radiation from the reactor vessel. Reference [12] discusses the radiation transferred between parallel plates through a gray gas. Figure 5.8 is a graph that shows the effective heat transfer between two blackbodies with respect to the optical thickness of the slab of gray gas between them. The optical thickness for the steam in the gap is one half of its emissivity. This graph is equally valid for determining the radiation heat transfer between gray surfaces. Therefore, in order to determine the effective heat transferred, the heat flux calculated using equation 5.3 is multiplied by the appropriate attenuation factor. This factor is found by using the corresponding total flux line of Figure 5.8 for the optical thickness of the water vapor in the gap.

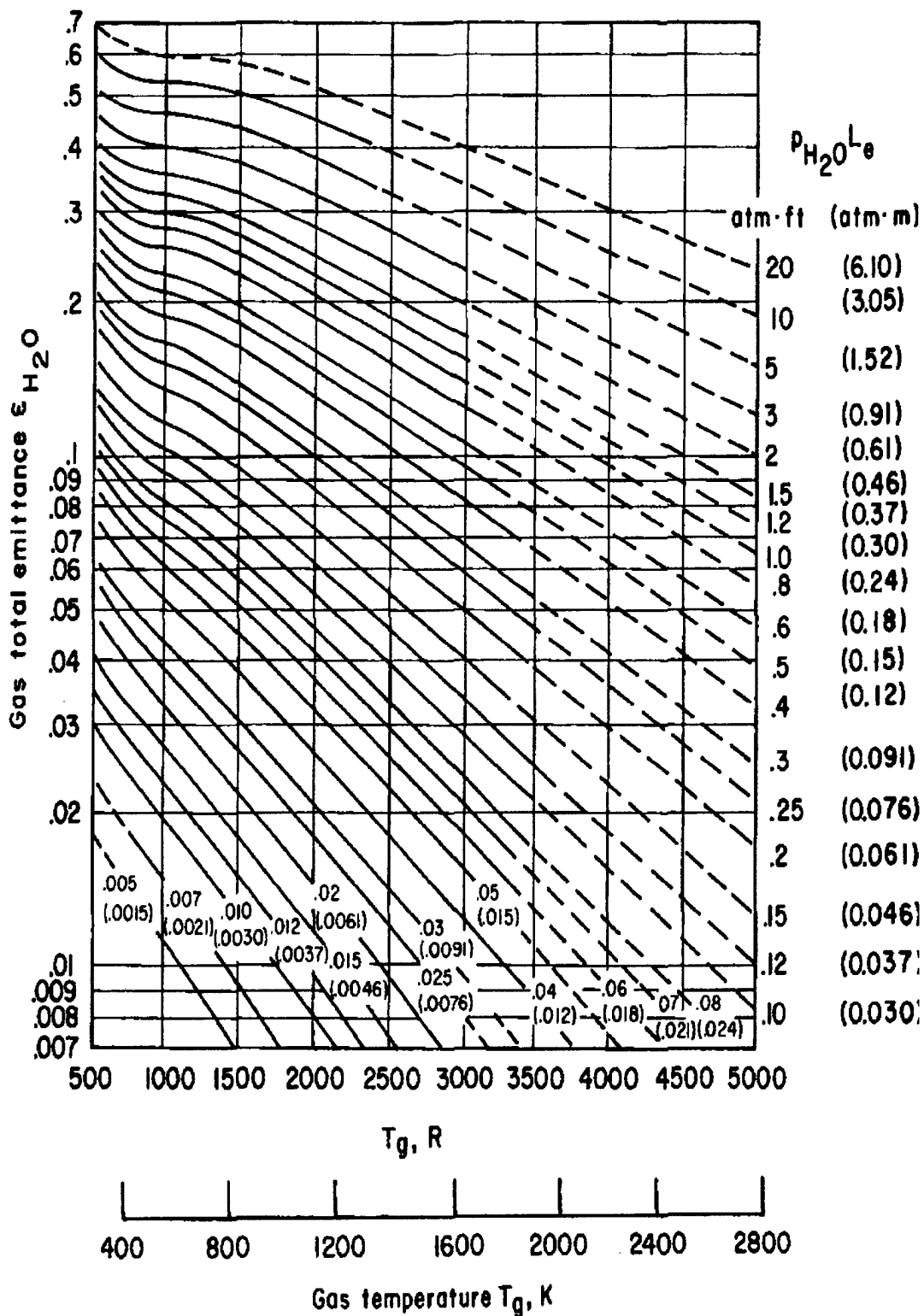


Figure 5.7 Total Emissivity of Water Vapor [11]

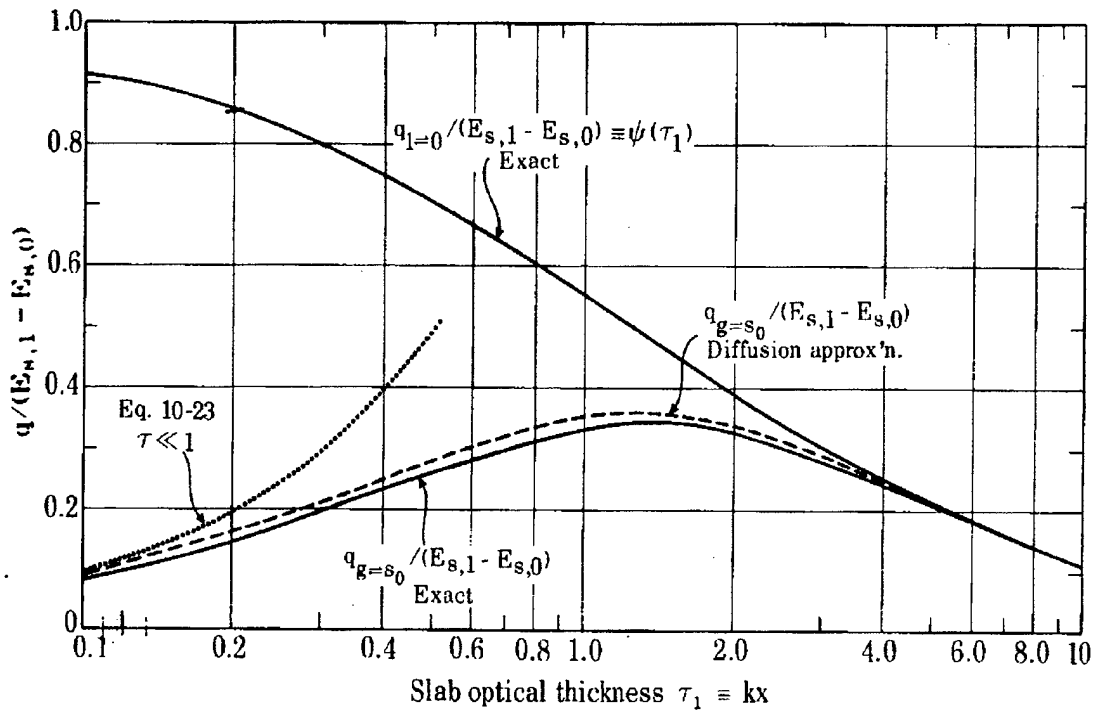


Figure 5.8 Net Radiative Flux through a Gray Gas [12]

As is discussed elsewhere in this report, the effect of steam on RCCS performance was found to be relatively small. The casualty of operators failing to refill the RCCS uses a different approach to perform the heat transfer analysis. The RCCS is assumed to stay at 100 °C until the system ceased to function at approximately three days. Instead of using analytical solutions to solve for the heat flux and then specifying the heat flux to the finite element program, a radiation heat transfer coefficient is determined. This heat transfer coefficient is used in the finite element model to determine the radiation heat flux off of the vessel. The two methods used for analyzing the radiation heat flux have very similar results. The second method used to calculate the radiation heat flux from the vessel to the cylinders is [7]:

$$q'' = h_r * (T_{\text{vessel}}^2 - T_{\text{RCCS}}^2) * (T_{\text{vessel}} - T_{\text{RCCS}}) \quad (5.4)$$

Where:

q'' : Heat Flux (W/m^2)

T_{vessel} : Reactor Vessel Temperature

T_{RCCS} : RCCS Cylinder Temperature

h_r : Radiative Heat Transfer Coefficient

$$h_r = \frac{\sigma}{1/\varepsilon_1 + (A_1/A_2)(1/\varepsilon_2 - 1)} \quad (5.5)$$

Where:

σ : Stephan-Boltzmann Constant ($5.67 \cdot 10^{-8} \text{ W / m}^2 \text{ K}^4$)

ε_1 : Higher Temperature Surface Emissivity

ε_2 : Lower Temperature Surface Emissivity

A_1 : Higher Temperature Surface Area (m^2)

A_2 : Lower Temperature Surface Area (m^2)

5.4 Natural Convection between Vessel and RCCS

The natural convection heat transfer between the reactor vessel and the surrounding water-filled RCCS cylinders is affected by the composition of the mixture in the gap between the two surfaces. The RCCS system consists of 45 cylinders and the gaps between cylinders is bridged with a thermal shield. The natural circulation in this region is driven by the vessel being significantly higher than the RCCS cylinder. The resulting flow is up along the reactor vessel surface then down along the RCCS. In order to enable an analytical solution to be found the RCCS is modeled as a flat cylindrical shell.

There are two non-dimensional numbers that are important when evaluating natural convection between two surfaces: Rayleigh number and Nusselt number. The Rayleigh number is the non-dimensional relation between buoyancy and viscosity forces. The Nusselt number is the non-dimensional heat transfer. For a cylinder surrounded by a cylindrical shell the equations for the Rayleigh and Nusselt numbers are [11]:

$$Ra = \frac{g \cdot B(T_i - T_o)L^3}{\nu \cdot \alpha} \quad (5.6)$$

$$Nu = 0.09 Ra^{.278} (D_o/D_i)^{.34+.329 Do/Di} (L/H)^{0.122} (D_i/L) \ln(D_o/D_i) \quad (5.7)$$

Where:

Ra: Rayleigh number

g: gravitational constant (9.81 m/s^2)

T_i : temperature of inner surface

T_o : temperature of outer surface
 B : coefficient of thermal expansion
 L : characteristic length: $L = (D_o - D_i)/2 = .8$ meters
 ν : kinematic viscosity
 H : height of cylinders

The validity for Nusselt number equation is valid for the ranges: $Ra < 10^6$, $2 < (D_o/D_i) < 15$, $1 < H/L < L$. For the problem of convection heat transfer between the reactor vessel and the RCCS cylinders the following relations apply:

$$\begin{aligned}
 H/L &= 12.5 \\
 D_o/D_i &= 1.27 \\
 Ra &= 5 \cdot 10^8 - 2 \cdot 10^9
 \end{aligned}$$

The Rayleigh number is well beyond the valid range for the Nusselt number equation. The range is exceeded due to the fact that the gap between the two surfaces is relatively small; which results in a small difference in surface area. The values for convective heat transfer were using this method were much lower than those calculated by the Eskom. A more appropriate way to solve for the heat transfer is treat the two surfaces as parallel plates, due to the small difference in surface area. The Nusselt number for this case is [10]:

$$Nu = .22 \left(\frac{Pr}{0.2 + Pr} Ra \right)^{0.28} \left(\frac{H}{L} \right)^{-1/4} \quad (5.8)$$

Where:

Pr : Prandtl number

The Prandtl number is determined from the physical properties of the mixture in the gap between the two surfaces. All properties for this mixture are evaluated at the mean temperature of the two surfaces. The RCCS is assumed to have a fixed temperature of 100 °C since this is the boiling point of the water in the RCCS. The Nusselt number is used to determine the heat transfer coefficient for the two surfaces then the natural circulation heat flux is [9]:

$$h_{conv} = \frac{Nu \cdot L}{k} \quad (5.9)$$

$$q_{conv} = h_{conv} \cdot (T_{vessel} - T_{RCCS}) \quad (5.10)$$

Where:

q_{conv} : convective heat flux between vessel and RCCS (W/m²)

k : thermal conductivity of mixture in gap

h_{conv} : convective heat transfer coefficient

The convection and radiation heat fluxes are combined with the conduction heat flux to give the total heat flux from the vessel to the RCCS. This heat flux is used as an input for the vessel region defined in the HEATING finite element analysis. It is then possible to determine the response of the PBMR for various mixtures of steam and air in the gap between the vessel and the RCCS cylinders by varying the physical properties used in the calculations.

5.5 Effect of Steam on the Heat Transfer out of the Reactor Vessel

Steam is not transparent to thermal radiation and therefore reduces the amount of heat transfer out of the vessel. This reduction in heat transfer results in elevated core and vessel temperatures. This section details the effect of changing the percentage of steam in the gap on the total heat flux out of the vessel. The heat flux out of the vessel is calculated for ten different values of vessel temperature. All of the physical and thermal properties of the mixture in the gap were found using the average temperature of the vessel and the RCCS.

Table 5.3 shows the conduction heat flux values. Table 5.4 contains the values for convective heat flux. Table 5.5 lists the radiation heat flux between the vessel and the RCCS. The values for the combined heat flux are given in Table 5.6. The values of heat flux are determined for steam with concentrations of: 100, 75, 50, 25, and 0 percent.

Table 5.3 Conduction Heat Flux out of Vessel

	Concentration of Steam				
	1	0.75	0.5	0.25	0
Temperature (°C)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)
279	6.46	6.86	7.26	7.659	8.059
350	10.333	10.851	11.37	11.888	12.406
450	16.29	16.935	17.579	18.224	18.869
550	23.413	24.065	24.717	25.369	26.021
650	31.543	32.104	32.665	33.226	33.787
750	40.643	41.025	41.408	41.79	42.172
850	51.047	51.047	51.047	51.047	51.047
950	62.445	61.944	61.443	60.942	60.441
1050	75.154	73.945	72.735	71.526	70.317
1150	89.38	87.259	85.139	83.019	80.898

Table 5.4 Convection Heat Flux out of Vessel

	Concentration of Steam				
	1	0.75	0.5	0.25	0
Temperature (°C)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)
279	288.767	293.508	298.27	302.869	307.127
350	431.07	437.102	443.244	449.264	454.933
450	653.353	658.724	664.337	669.885	675.052
550	885.068	888.266	891.926	895.701	899.216
650	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03
750	1.36E+03	1.36E+03	1.35E+03	1.35E+03	1.35E+03
850	1.63E+03	1.61E+03	1.60E+03	1.58E+03	1.56E+03
950	1.89E+03	1.86E+03	1.83E+03	1.81E+03	1.78E+03
1050	2.24E+03	2.18E+03	2.12E+03	2.06E+03	2.00E+03
1150	2.72E+03	2.59E+03	2.47E+03	2.34E+03	2.21E+03

Table 5.5 Radiation Heat Flux out of Vessel

	Concentration of Steam				
	1	0.75	0.5	0.25	0
Temperature (°C)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)
279	2.04E+03	2.09E+03	2.17E+03	2.30E+03	2.55E+03
350	3.65E+03	3.74E+03	3.88E+03	4.11E+03	4.56E+03
450	7.06E+03	7.24E+03	7.50E+03	7.94E+03	8.83E+03
550	1.22E+04	1.25E+04	1.30E+04	1.38E+04	1.53E+04
650	1.97E+04	2.01E+04	2.09E+04	2.21E+04	2.46E+04
750	2.99E+04	3.07E+04	3.18E+04	3.37E+04	3.74E+04
850	4.37E+04	4.48E+04	4.64E+04	4.92E+04	5.46E+04
950	6.17E+04	6.32E+04	6.55E+04	6.94E+04	7.71E+04
1050	8.47E+04	8.68E+04	9.00E+04	9.53E+04	1.06E+05
1150	1.14E+05	1.16E+05	1.21E+05	1.28E+05	1.42E+05

Table 5.6 Total Heat Flux out of Vessel

	Concentration of Steam				
	1	0.75	0.5	0.25	0
Temperature (°C)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)	Heat Flux (w/m ²)
279	2.34E+03	2.39E+03	2.48E+03	2.61E+03	2.87E+03
350	4.09E+03	4.19E+03	4.33E+03	4.57E+03	5.03E+03
450	7.73E+03	7.91E+03	8.18E+03	8.63E+03	9.52E+03
550	1.31E+04	1.34E+04	1.39E+04	1.47E+04	1.62E+04
650	2.08E+04	2.13E+04	2.20E+04	2.33E+04	2.57E+04
750	3.13E+04	3.21E+04	3.32E+04	3.51E+04	3.88E+04
850	4.54E+04	4.65E+04	4.81E+04	5.08E+04	5.62E+04
950	6.36E+04	6.51E+04	6.74E+04	7.13E+04	7.89E+04
1050	8.70E+04	8.90E+04	9.22E+04	9.74E+04	1.08E+05
1150	1.16E+05	1.19E+05	1.23E+05	1.30E+05	1.44E+05

5.6 Structural Analysis of Reactor Vessel

Elevated vessel temperatures which exceed the design temperature limit can result in damage to the vessel. The PBMR vessel is made out of SA-508, which is mild steel. These material properties of mild steel vary with temperature. Specifically, the yield strength is greatly reduced above 600 °C. Table 5.7 is a summary of the mechanical and thermal properties of mild steel.

Table 5.7 Mechanical and Thermal Properties of Mild Steel

Temp °C	Yield Stress (Mpa)	Young's Modulus (Cpa)	Poisson's Ratio	Thermal Exp. Coefficient (10 ⁻⁶ / °C)
0	290	200	0.3	10
100	260	200	0.3	11
300	200	200	0.3	12
450	150	150	0.3	13
550	120	110	0.3	14
600	110	88	0.3	14
720	9.8	20	0.3	14
800	9.8	20	0.3	15
1550	0.98	0.2	0.3	

The highest temperature on the modeled vessel is found at its vertical center. This region is called the beltline. As the temperature increases the modulus of elasticity and yield strength are reduced. If the vessel temperature is high enough the following sequence of events could occur:

1. Temperature increases due to impairment of the RCCS.
2. Yield strength decreases and plastic deformation starts and can lead to a rupture in the beltline region.
3. The rupture results in the internal pressure equalizes with external pressure. Citadel over-pressure system actuates removing the helium from the reactor cavity and PCU space. The Citadel consists of the reactor cavity and PCU spaces.

4. Pressure stress drops rapidly. The weight of the upper vessel head and vessel cylindrical shell create a compressive stress:
 - If the compressive stress exceeds the yield stress the vessel will plastically deform around the point of fracture.
 - This compressive stress can cause the vessel to buckle if it exceeds the critical buckling stress of the vessel.
5. Catastrophic failure of the vessel can result in mechanical/thermal shock and/or chemical attack of the fuel pebbles.
6. Damage to fuel causes fission products to be released to the reactor cavity.
7. Fission products escape the citadel via the pressure relief system as compartment temperatures rise and cause cavity pressure to increase.

The reactor vessel is a cylindrical with elliptical upper and lower heads. The internal pressure following a reactor scram is 4.8 MPa. As the core temperature increases the pressure will rise due to expansion of the helium gas. The maximum vessel stress is calculated from a combination of circumferential and tensile stresses. The equations that determine the pressure stress in the vessel are [15].

$$\sigma_{\theta\theta} = \frac{a \times p}{h} \quad (5.11)$$

$$\sigma_{xx} = \frac{a \times p}{2 \times h} \quad (5.12)$$

$$\sigma_{\max} = \frac{\sqrt{3}}{2} \frac{a \times p}{h} \quad (5.13)$$

Where:

$\sigma_{\theta\theta}$: circumferential stress

σ_{xx} : axial stress

h: thickness

a: average radius

p: pressure

The maximum stress for the vessel is then found and compared to the yield strength. If the maximum stress exceeds the yield strength of the vessel material plastic deformation occurs and a rupture is possible. The internal pressure equalizes with the external pressure when the vessel ruptures.

Once the pressure equalizes, the stress in the wall becomes compressive due to the weight of the upper vessel. The compressive stress is found by dividing the weight of the upper portion of the vessel by the cross-sectional area of the shell. Using this method the maximum compressive stress is found to be 1.2 MPa. The vessel is then analyzed to determine if it will buckle. Reference [16] discusses cylindrical vessels in compression from an axial force. In order to determine the stress at which the buckling will occur we use the following equations [16]:

$$\sigma_{cr2} = \frac{3}{5} \cdot \frac{E \cdot h}{a \cdot \sqrt{3 \cdot (1 - \nu)}} \quad (5.14)$$

$$\sigma_{ult} = E \frac{0.6 \frac{h}{a} - 10^{-7} \frac{a}{h}}{1 + 0.004 \frac{E}{\sigma_{yield}}} \quad (5.15)$$

Where:

σ_{cr} : Critical Stress

σ_{ult} : Ultimate Stress

σ_{yield} : Yield Stress

ν : Poisson's Ratio

The critical stress equation is an analytical solution using the Euler method. Test results have shown that vessels in axial compression fail at stress levels well below the critical stress. The ultimate stress equation is based on empirical results, and more accurately predicts the vessel behavior. The effects of vessel temperature on the structural response of the PBMR are discussed further in the next chapter.

Chapter 6 Results of the Analysis

In order for probabilities of failure to be assigned to the proposed casualty the effects of the casualty are determined. The first casualty being analyzed is a rupture in a passive cooling system (RCCS) for the PBMR. This rupture generates steam due to the cooling water splashing on the hot reactor vessel. As discussed in chapter 5, steam in the gap reduces the amount of radiation heat transfer from the vessel to the RCCS. The second casualty analyzed was failure of the RCCS due to operator inaction. The operators failing to refill the RCCS casualty is shown to be a much more limiting case than steam blanketing casualty. This chapter summarizes the results of the heat transfer analysis for both casualties. Finally, the structural response of the vessel to the temperature transient is discussed.

6.1 Thermal Response Analysis Results

Steam Blanketing

This analysis section deals with the effect that steam blanketing of the reactor vessel has on its passive heat removal. The HEATING model of the PBMR was used to predict the response of the plant to various concentrations of water vapor in the gap between the reactor vessel and the RCCS cooling cylinders. The results of these analyses are shown in Figures 6.1-6.10. The water vapor concentrations used in the analysis are: 100, 75, 50, 25, and 0 percent.

The effect on the water vapor on the passive heat removal of the reactor vessel is not as great as initially expected. With the gap filled completely with steam the radiation heat flux is reduced by 20 percent. The presence of Steam results in a maximum vessel temperature increase of 32 °C above the maximum vessel temperature with only air present. Figure 6.11 is a graph of maximum core temperature versus time for all the different gap mixtures. Figure 6.12 is a graph of the vessel temperature with respect to time for all the data runs.

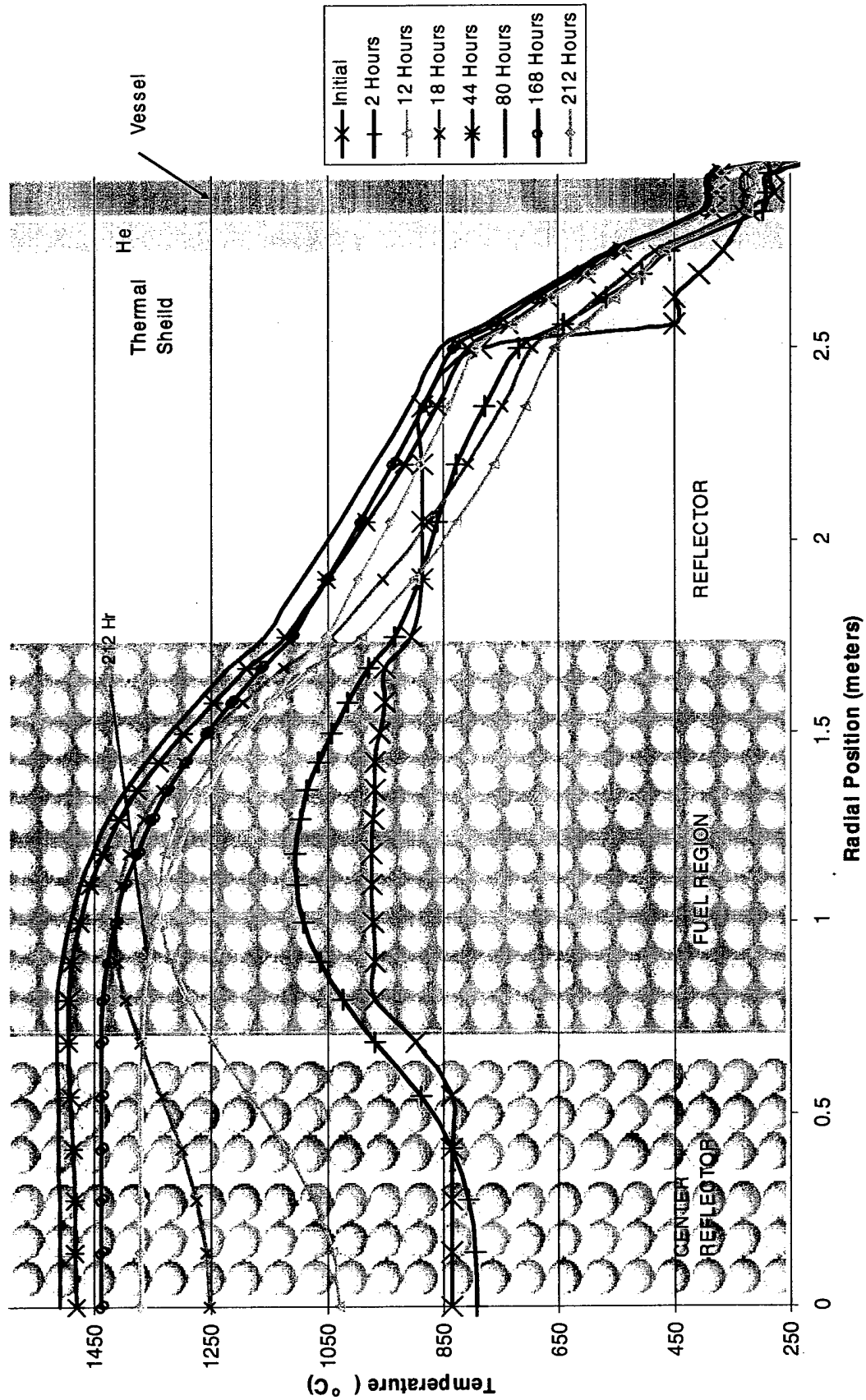


Figure 6.1 Radial Temp. Distributions w/ no Steam in the Annular Region

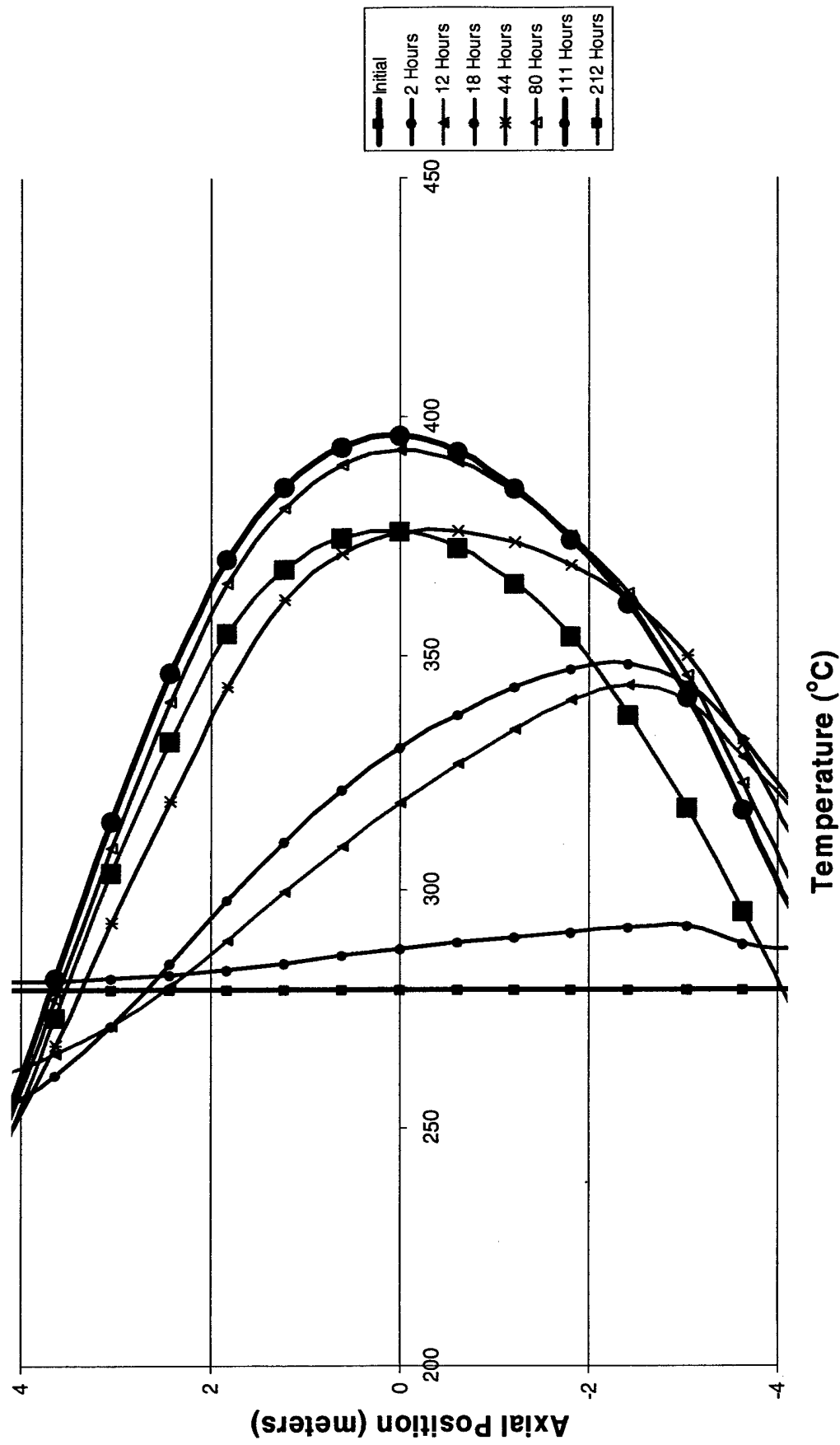


Figure 6.2 Axial Vessel Temp. Distributions w/ no Steam in the Annular Region

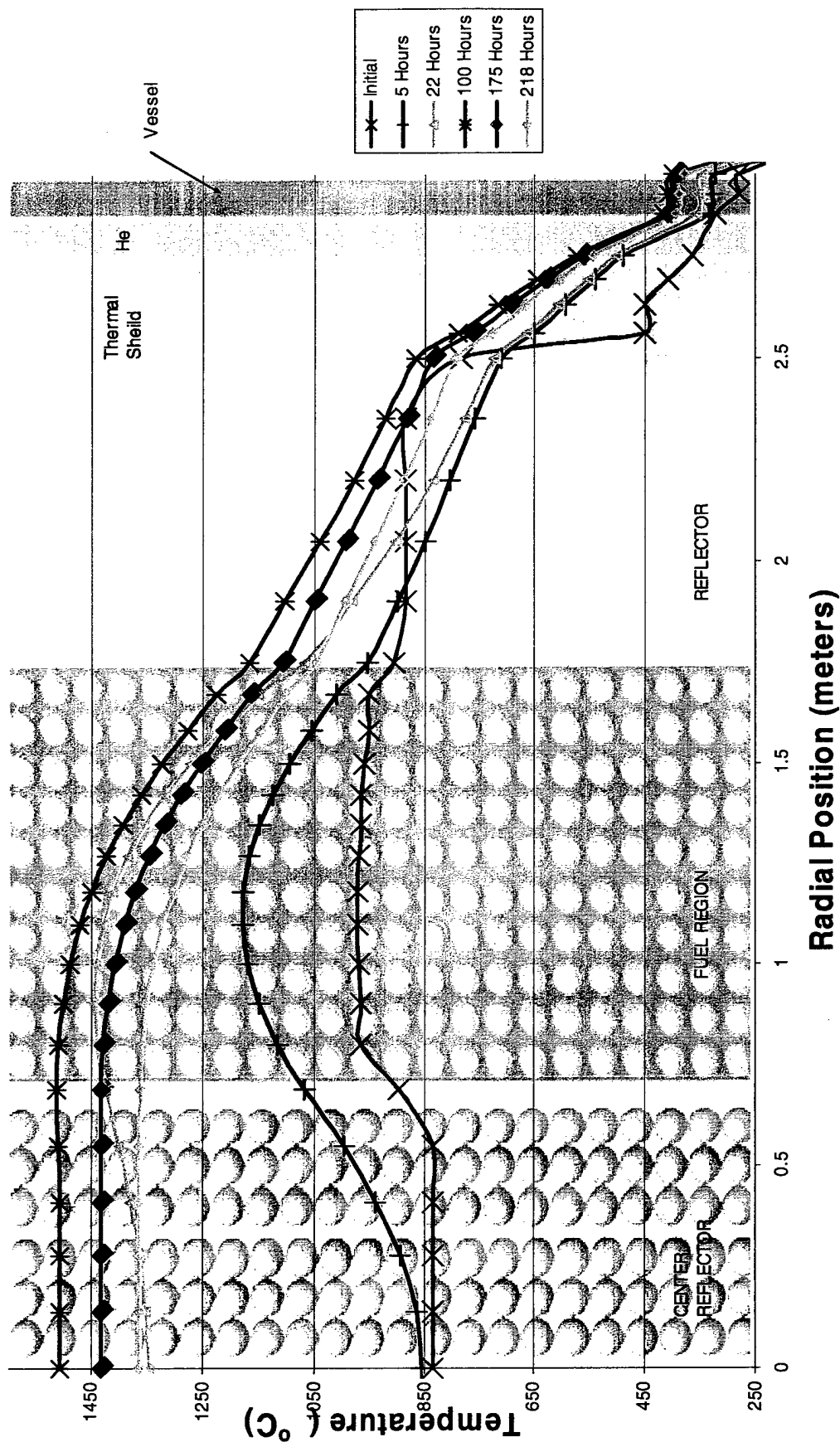


Figure 6.3 Radial Temp. Distributions w/ 25% Steam in the Annular Region

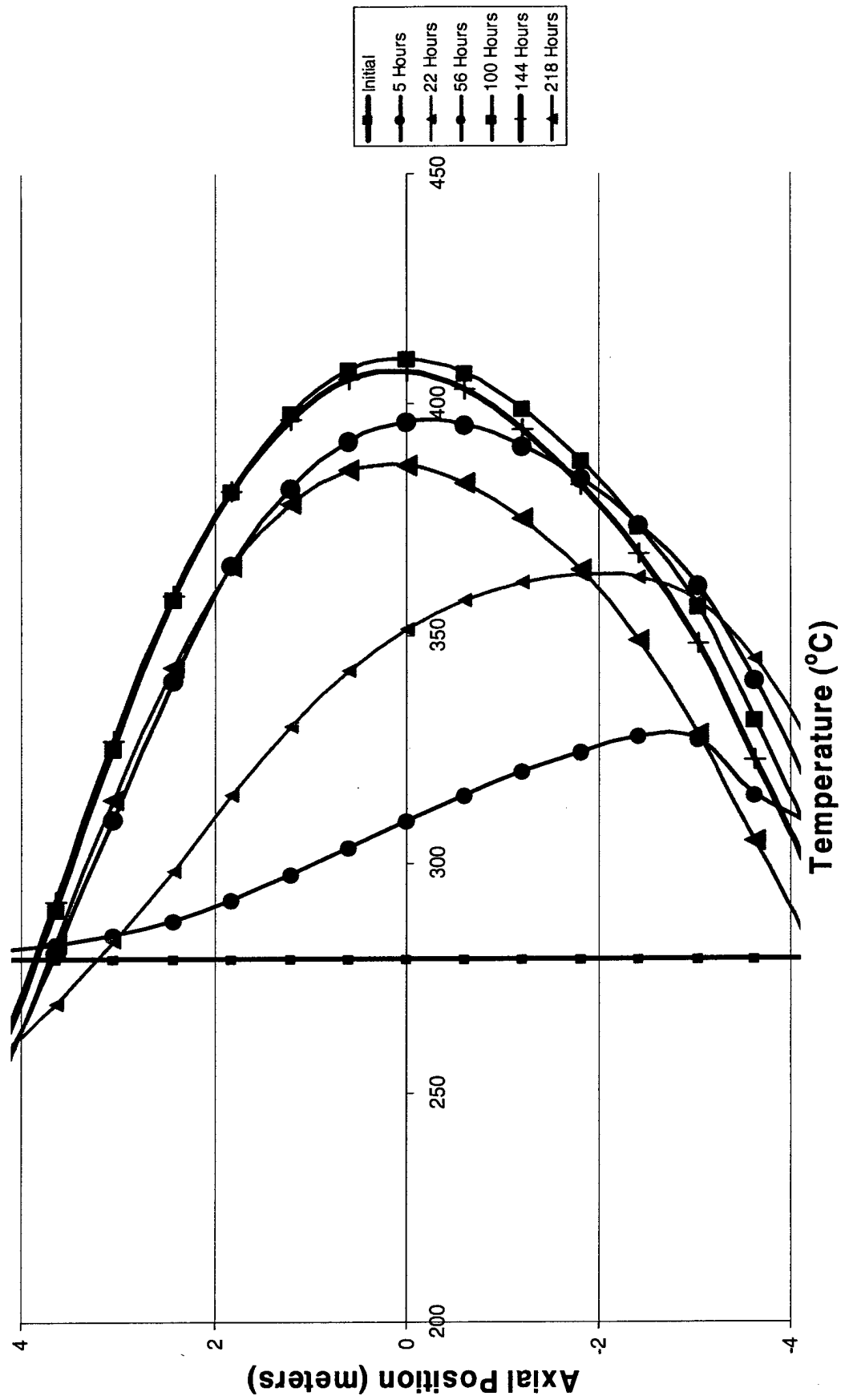


Figure 6.4 Axial Vessel Temp. Distributions w/ 25% Steam in the Annular Region

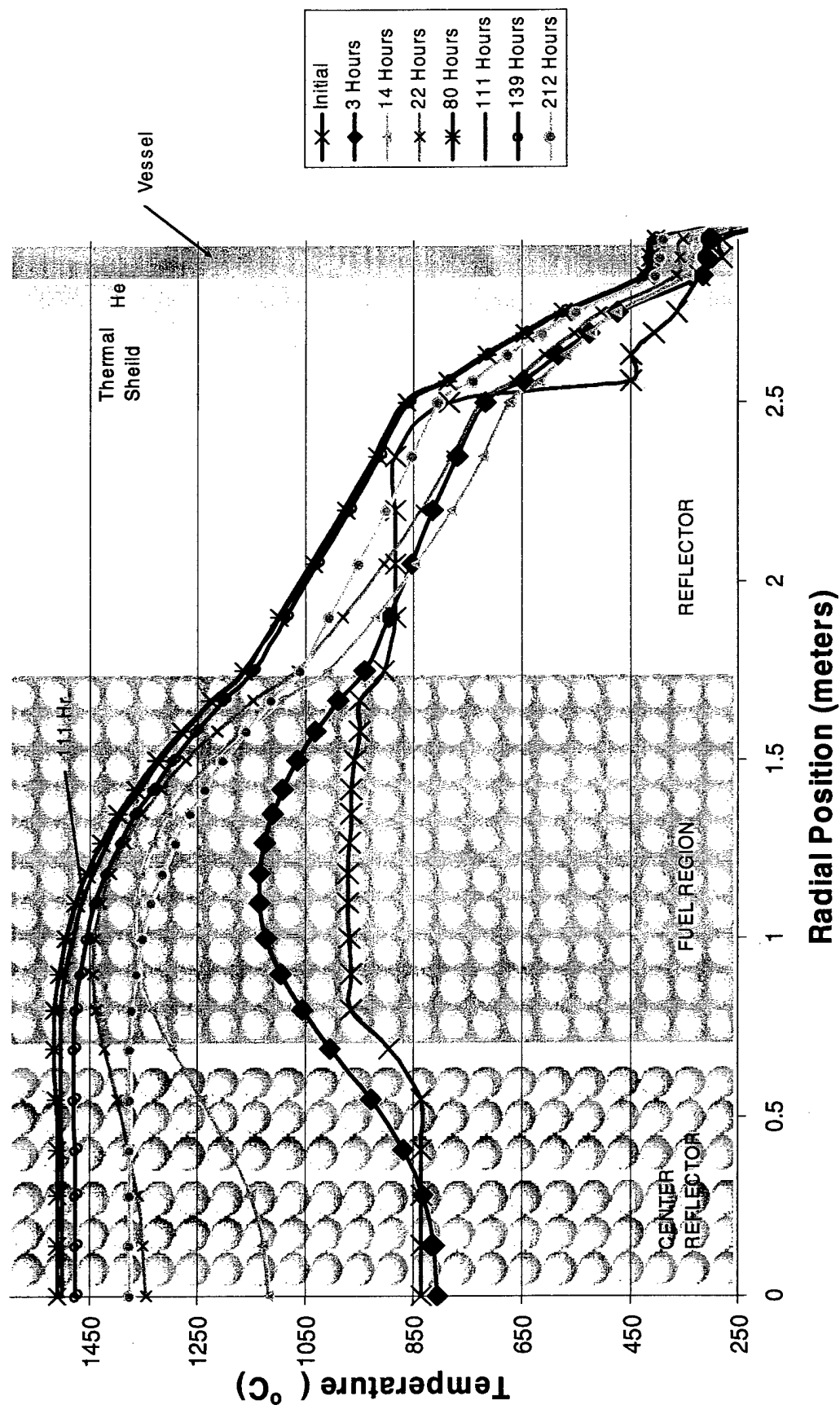


Figure 6.5 Radial Temp. Distributions w/ 50% Steam in the Annular Region

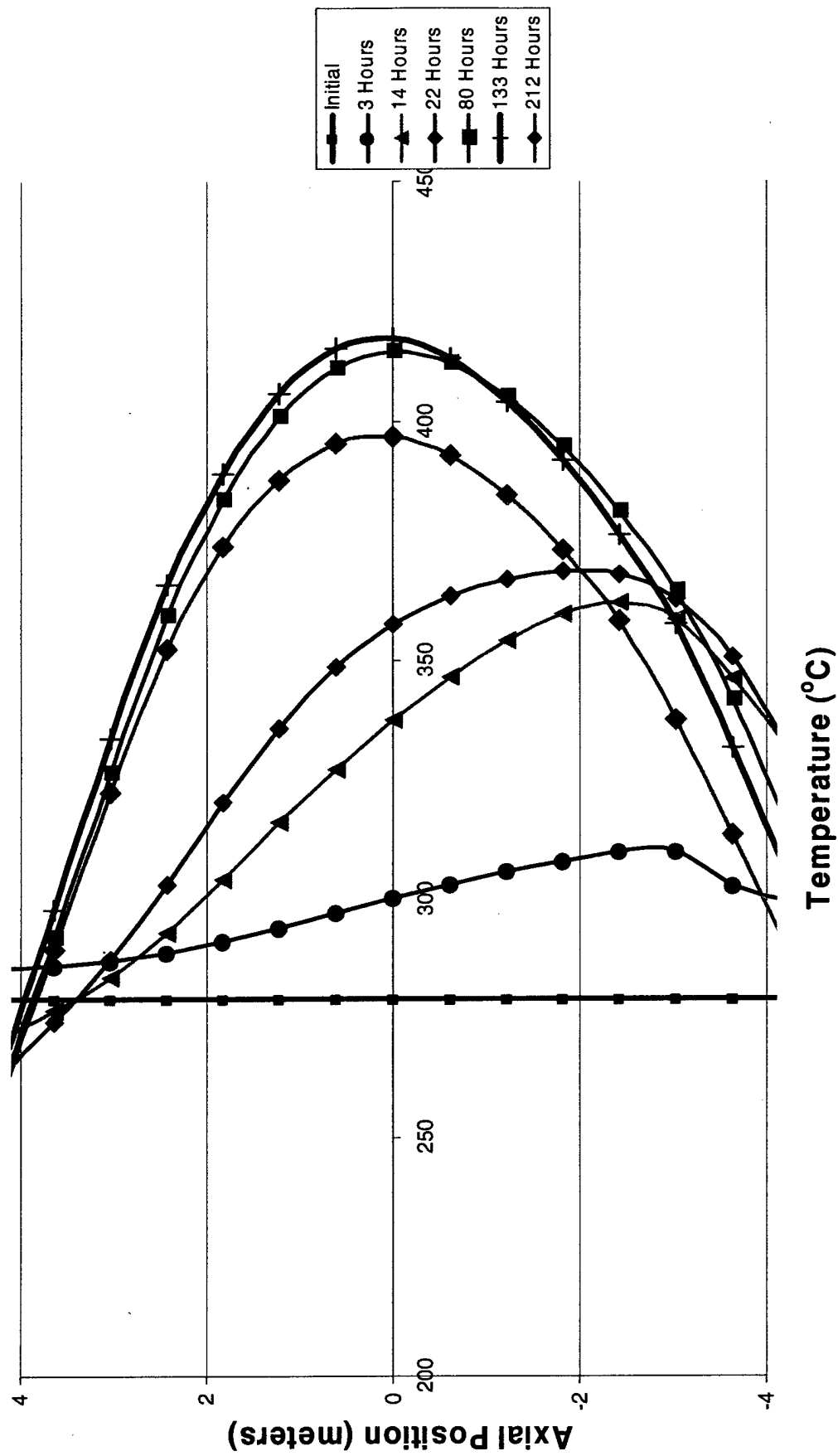


Figure 6.6 Axial Vessel Temp. Distributions w/ 50% Steam in the Annular Region

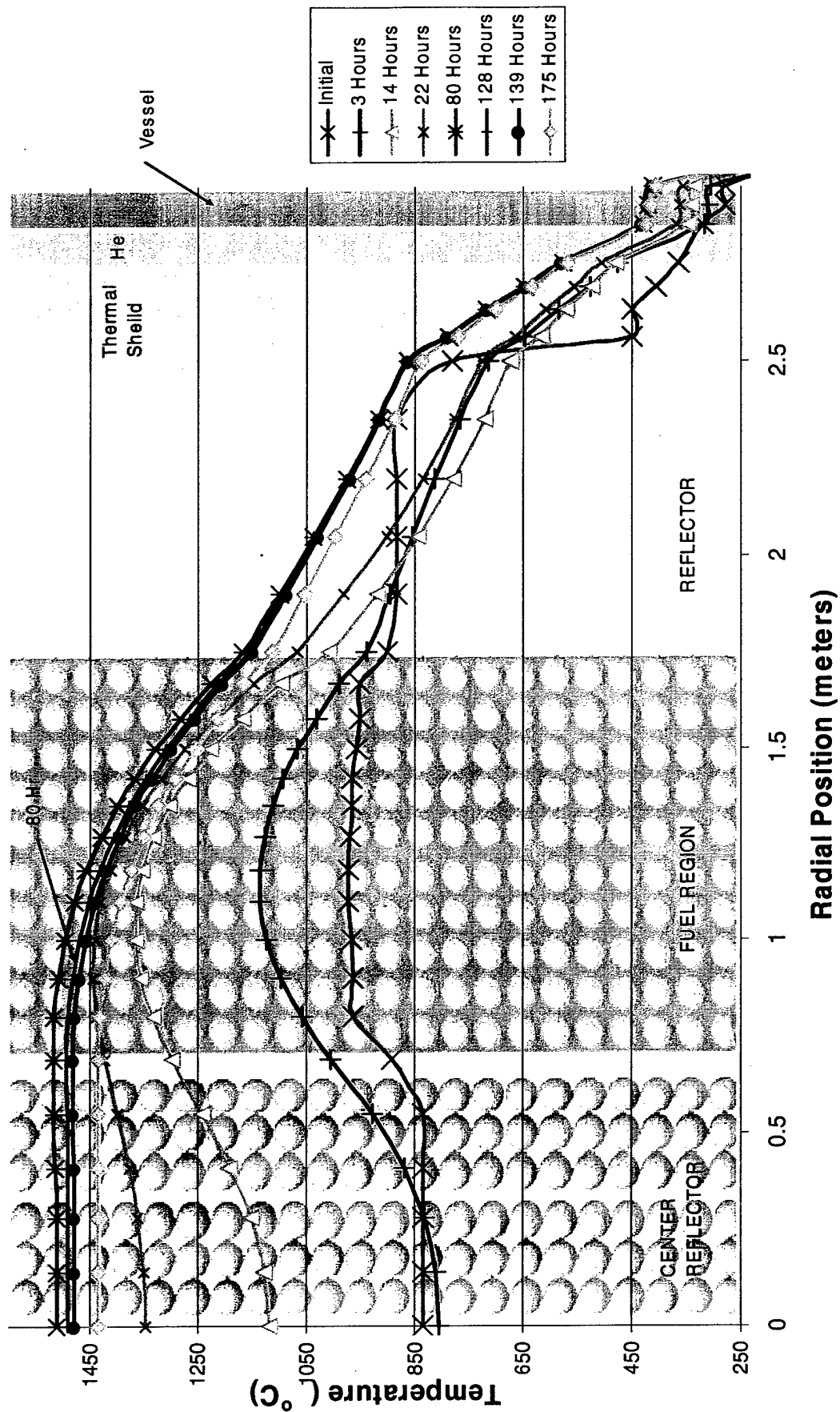


Figure 6.7 Radial Temp. Distributions w/ 75% Steam in the Annular Region

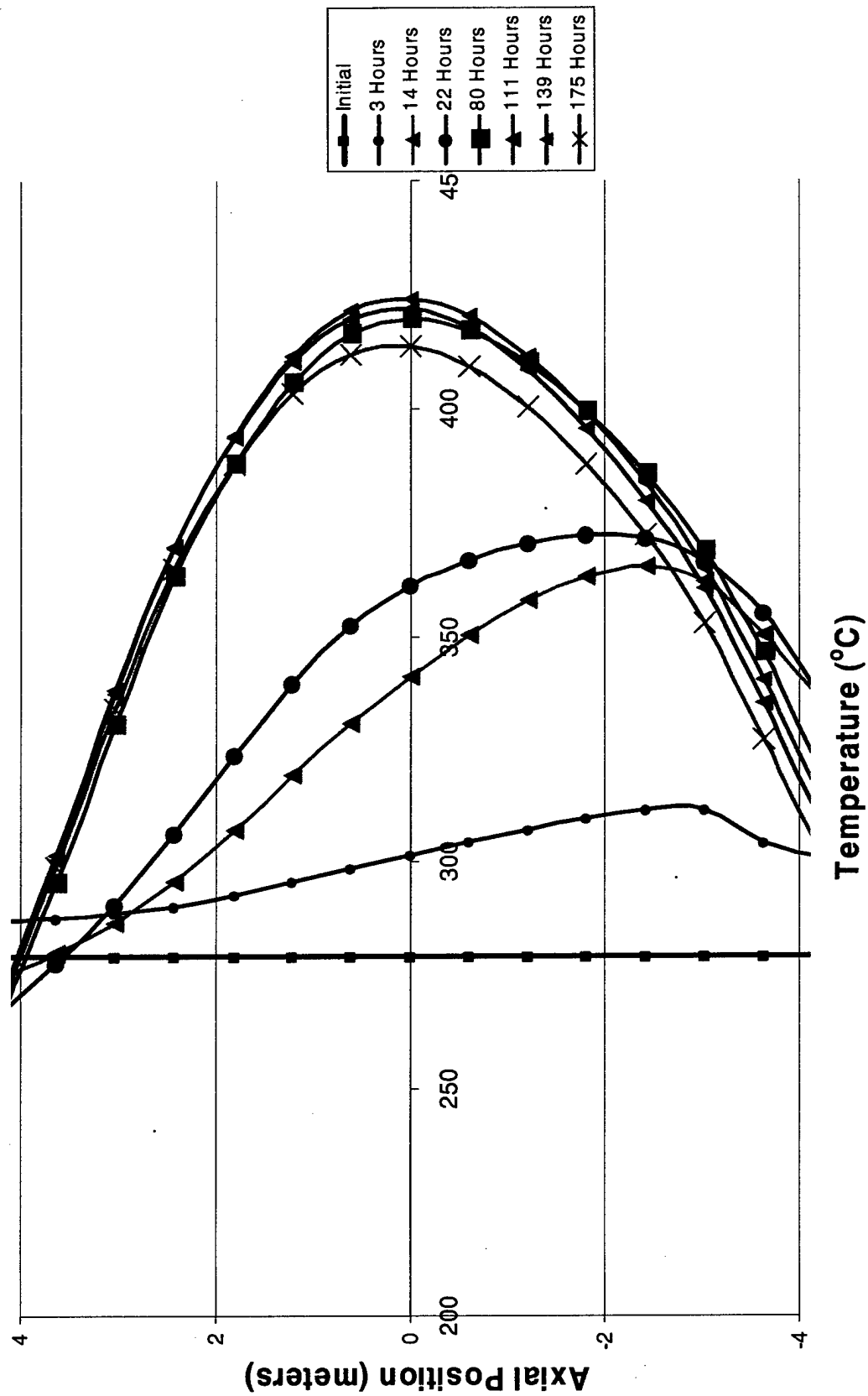


Figure 6.8 Axial Vessel Temp. Distributions w/ 75% Steam in the Annular Region

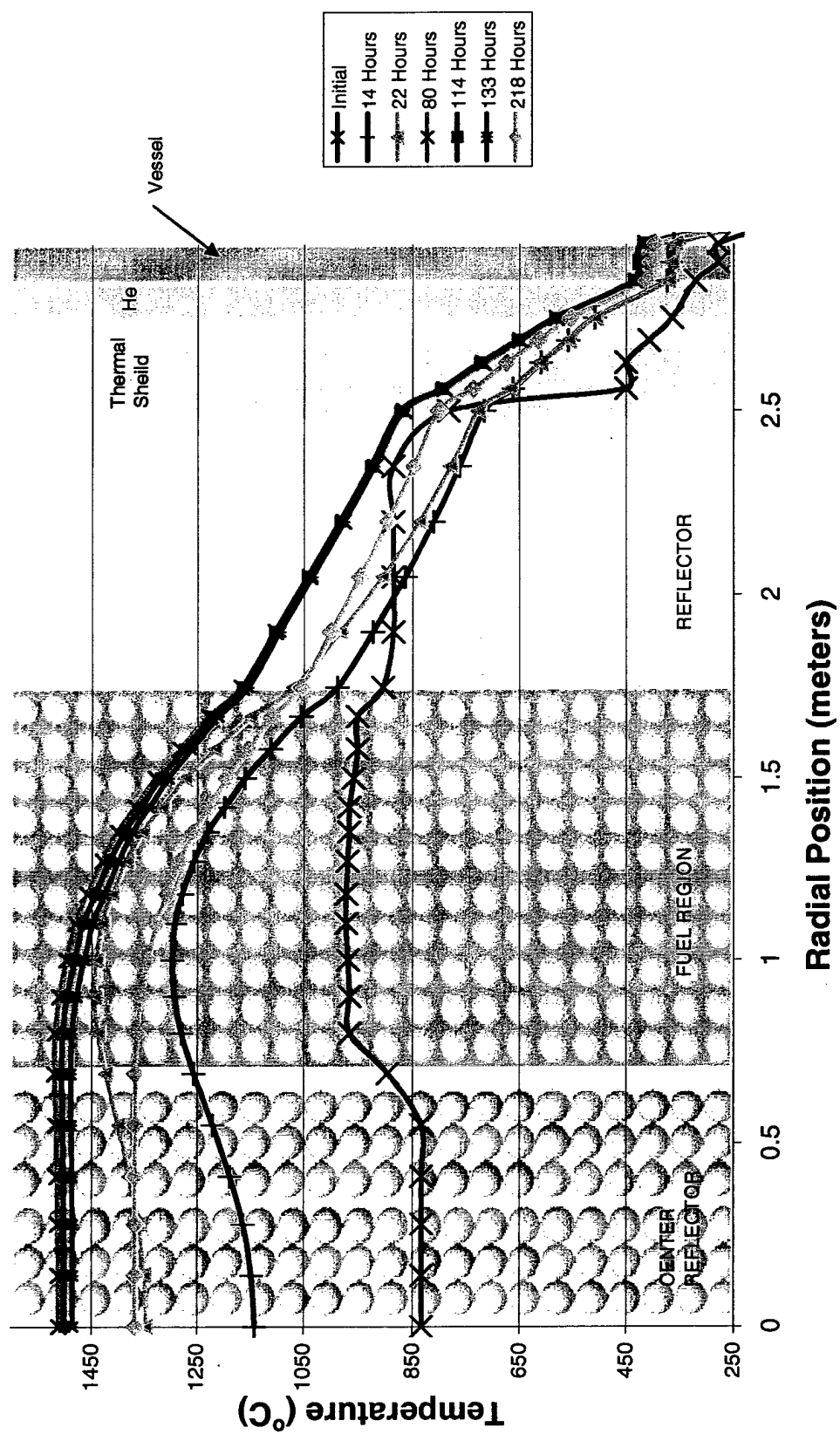


Figure 6.9 Radial Temp. Distributions w/ 100% Steam in the Annular Region

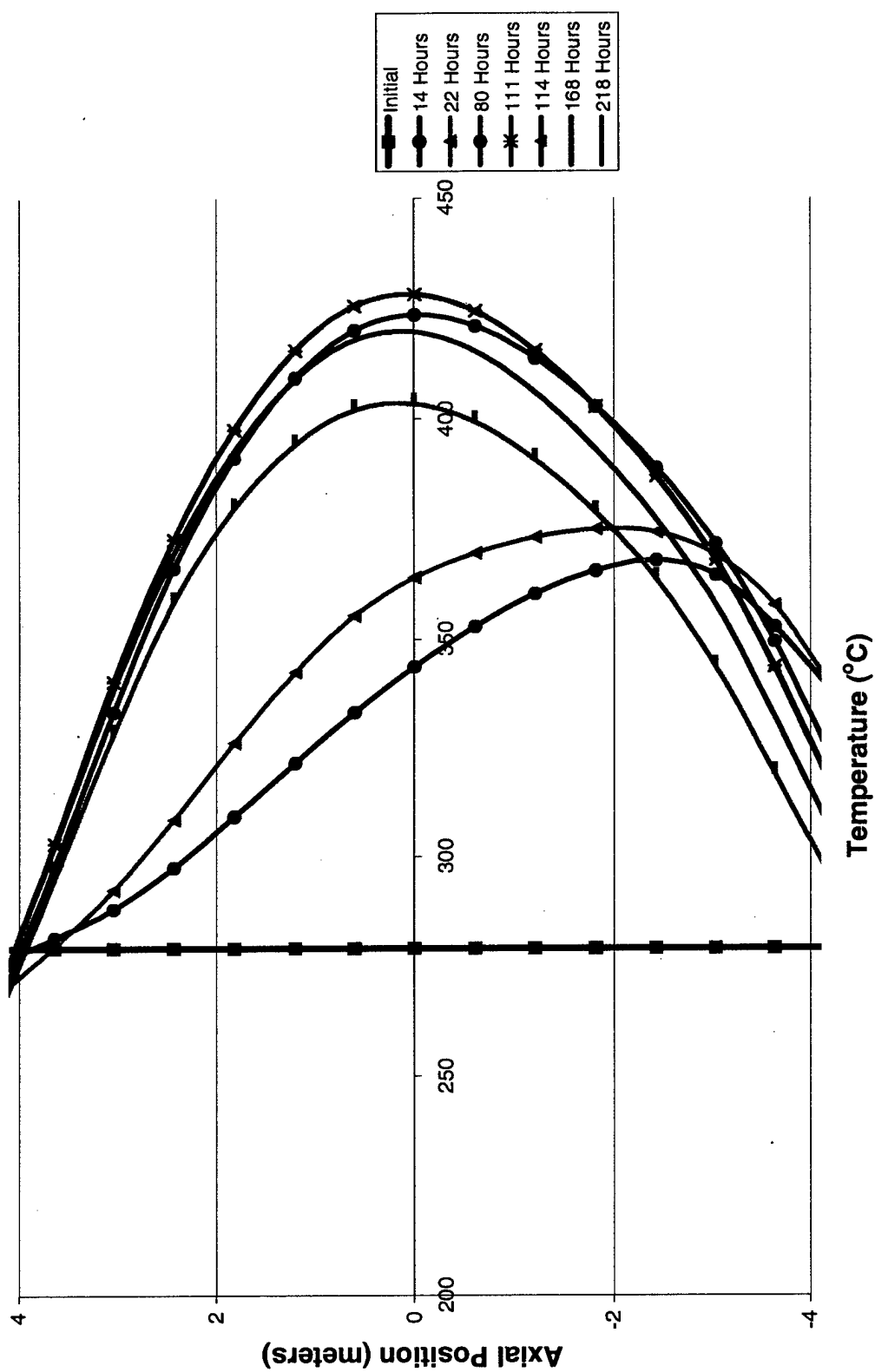


Figure 6.10 Axial Vessel Temp. Distributions w/ 100% Steam in the Annular Region

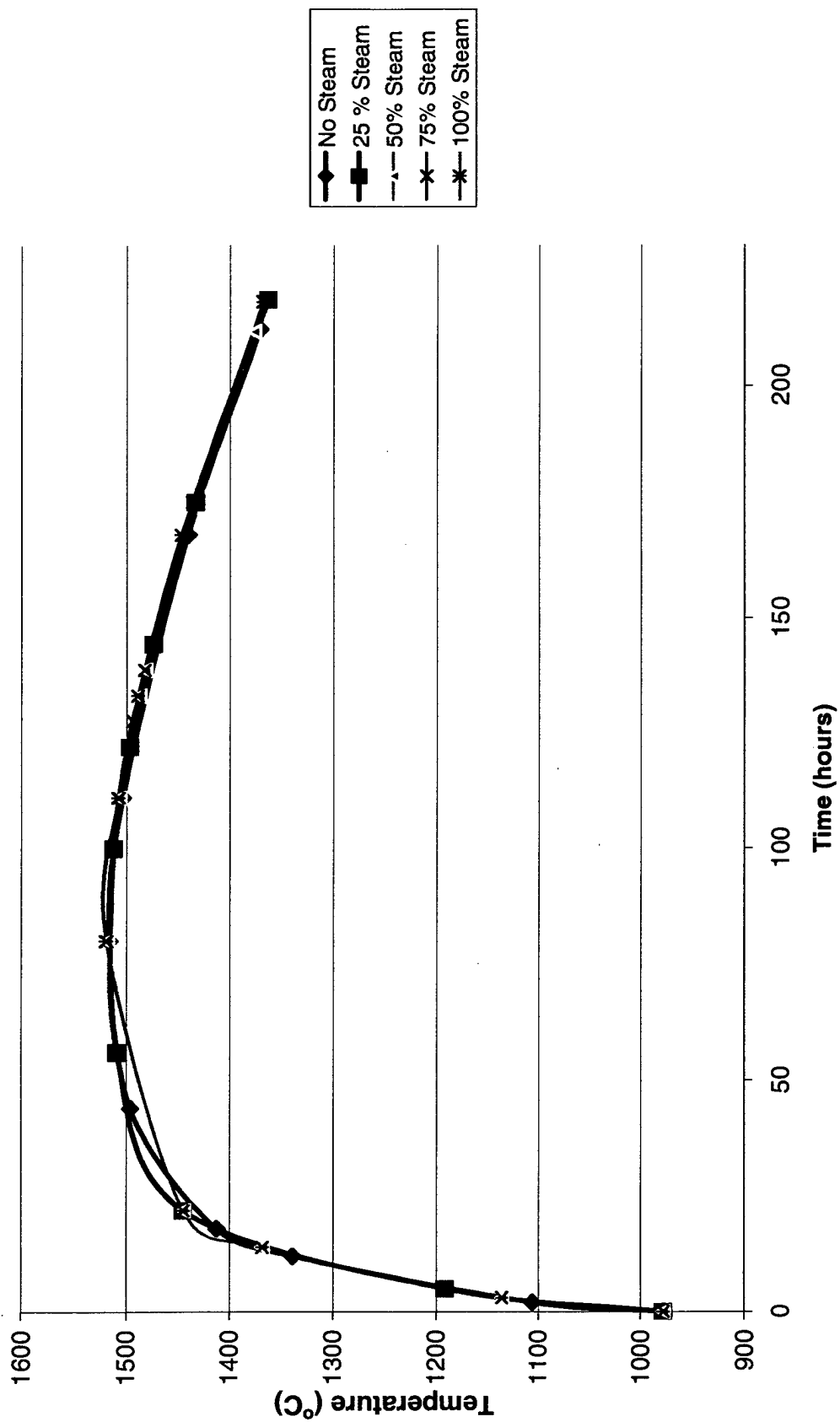


Figure 6.11 Maximum Core Temperatures vs. Time

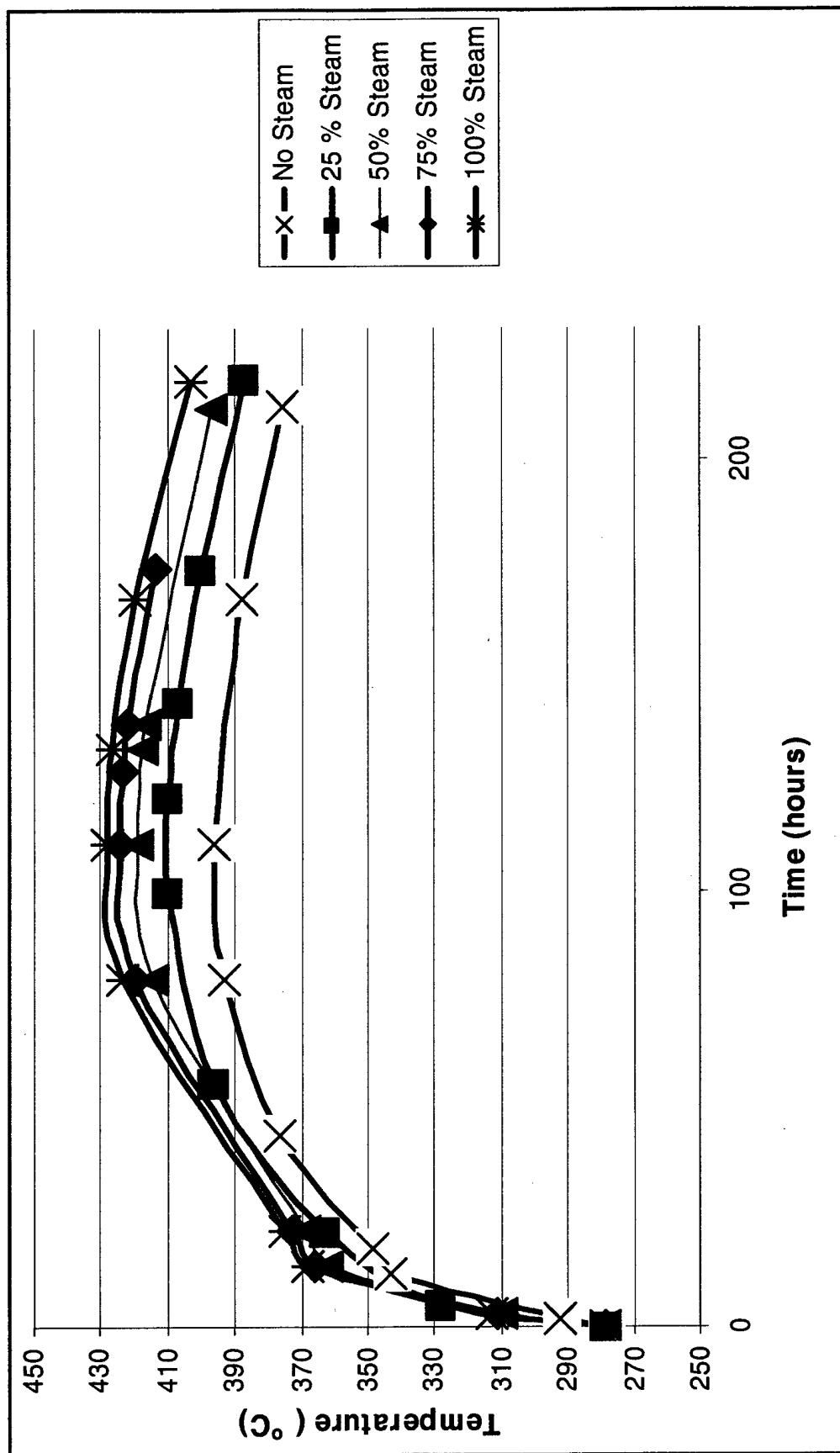


Figure 6.12 Maximum Vessel Temperatures vs. Time

RCCS Failure due to Operator Inaction

The effect of steam in the reactor cavity is shown to be relatively small. The second casualty more directly impairs the operation of the RCCS. The system will stop removing heat from the reactor cavity when a significant amount of the water in the tubes has been evaporated. Operators can prevent this failure by adding water to the system. Operators not taking the proper action could be the result of either operator error or lack of operator access to the facility.

This casualty assumes that no operator action is taken and the RCCS completely fails at three different times. The amount of time prior to RCCS failure is determined by comparing the heat flux transferred from the reactor vessel to the RCCS. This heat causes the water in the tubes to boil off. Therefore, various volumes of water boiling off are associated with the amount of time it took to evaporate them. At approximately 3 days 70 percent of the water is evaporated. The actual response of the system would be a gradual reduction in the rate of heat transfer. The exact time that the RCCS ceases to function is unknown. For this reason three different start times for the casualty are analyzed: 72 hours, 87 hours, and 100 hours.

First the response of the PBMR is determined assuming no failure of the RCCS. The cylinders of the RCCS are assumed to stay at 100 °C. At the start of the casualty the boundary conditions of the inner gap between the RCCS and the vessel are changed. The RCCS cylinders are allowed to heat up and conduct heat to the outer gap between the RCCS and the cavity wall. The material properties of the models RCCS region are modified to simulate the fact that heat from the vessel is conducted around the circumference of the RCCS cylinders and then radiated to the walls of the cavity. Figures 6.13-20 show the temperature responses of the reactor vessel and the entire reactor cavity model to the casualty for the four cases: RCCS functioning, failure at 72 hours, failure at 87 hours, and failure at 100 hours. Figure 6.21 is a graph of the maximum core and vessel temperatures for the four cases. We see that excessive vessel temperatures result in all the cases for a failed RCCS.

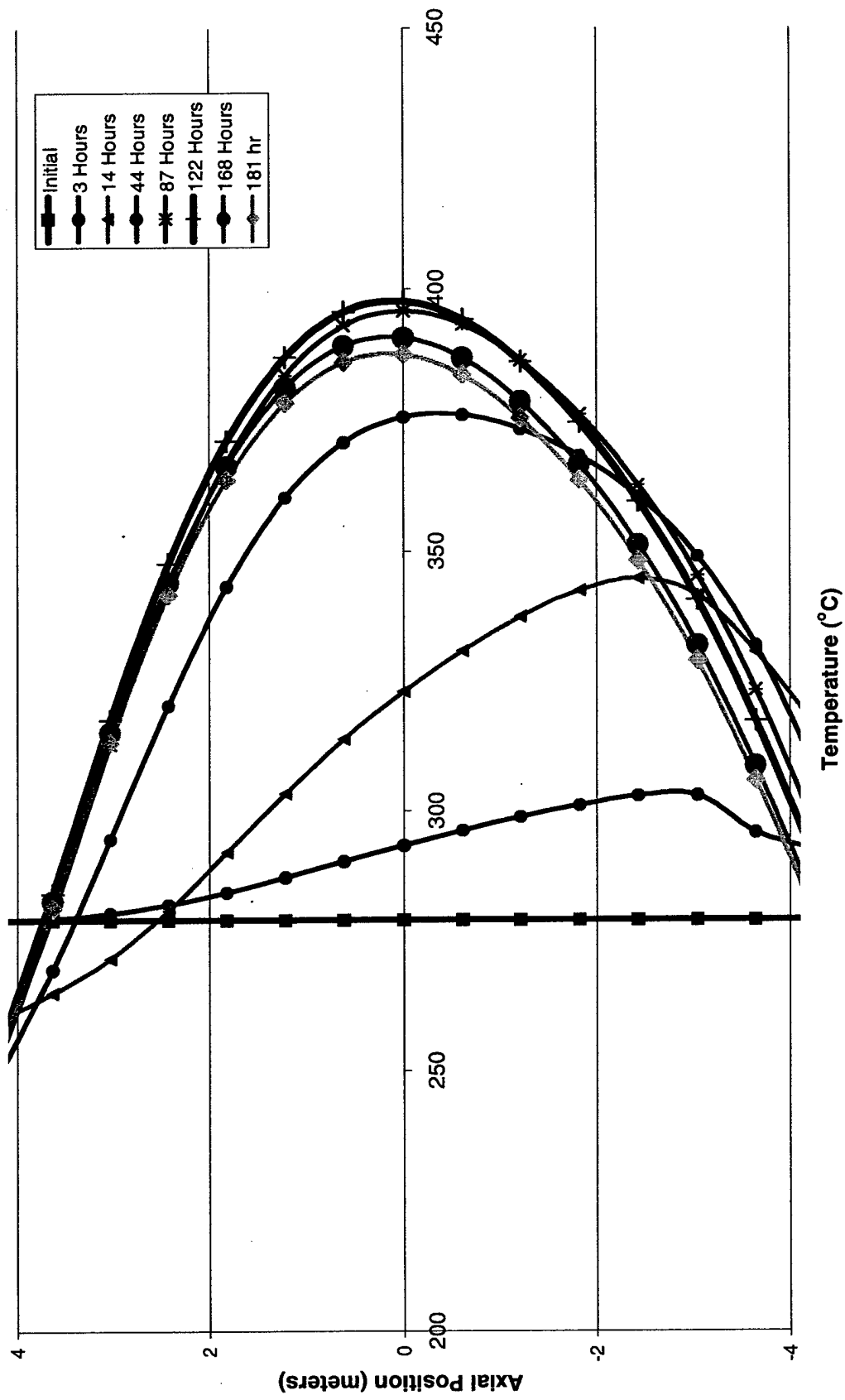


Figure 6.13 Vessel Temperature with RCCS Functioning

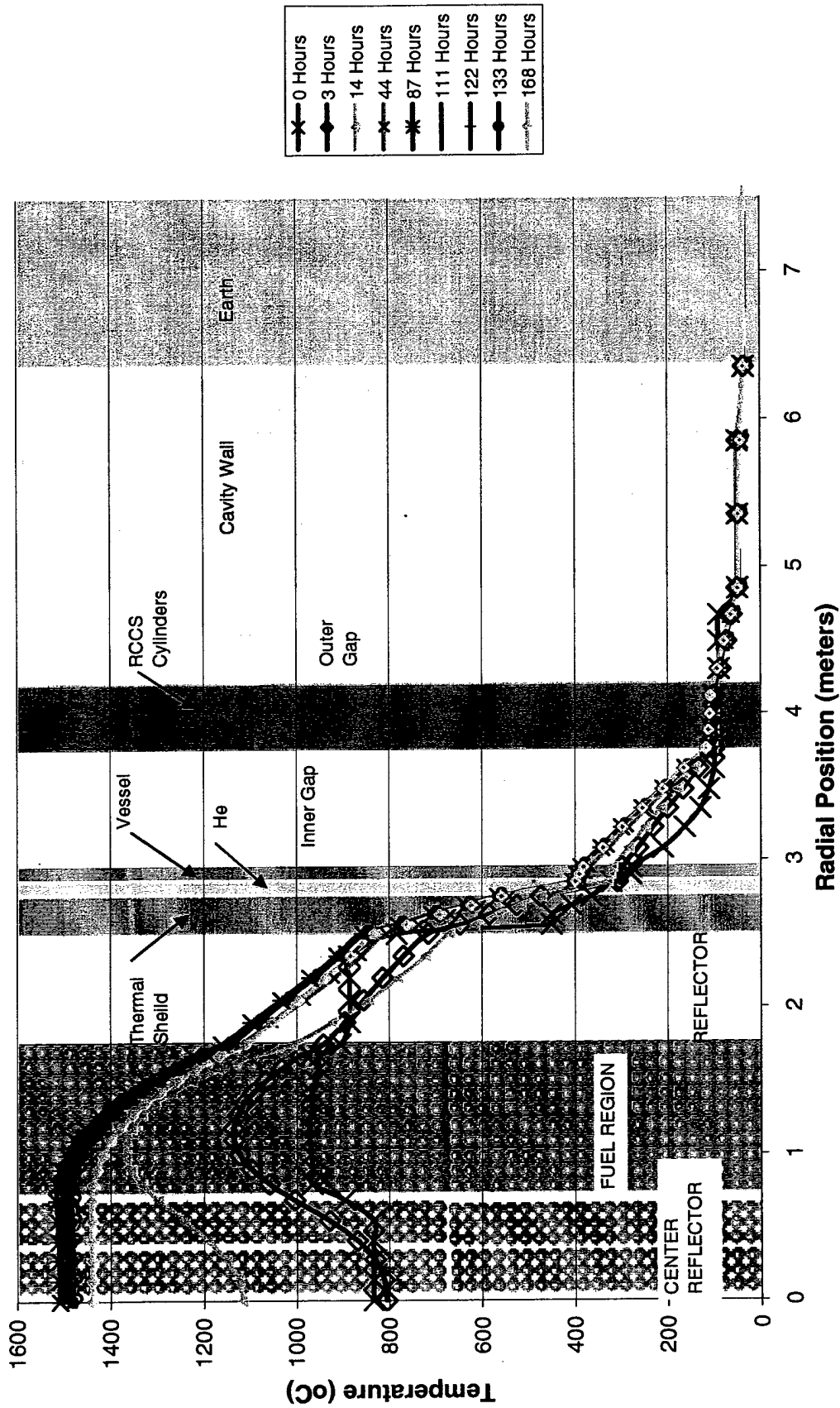


Figure 6.14 Reactor Cavity Radial Temperature Profile with RCCS Functioning

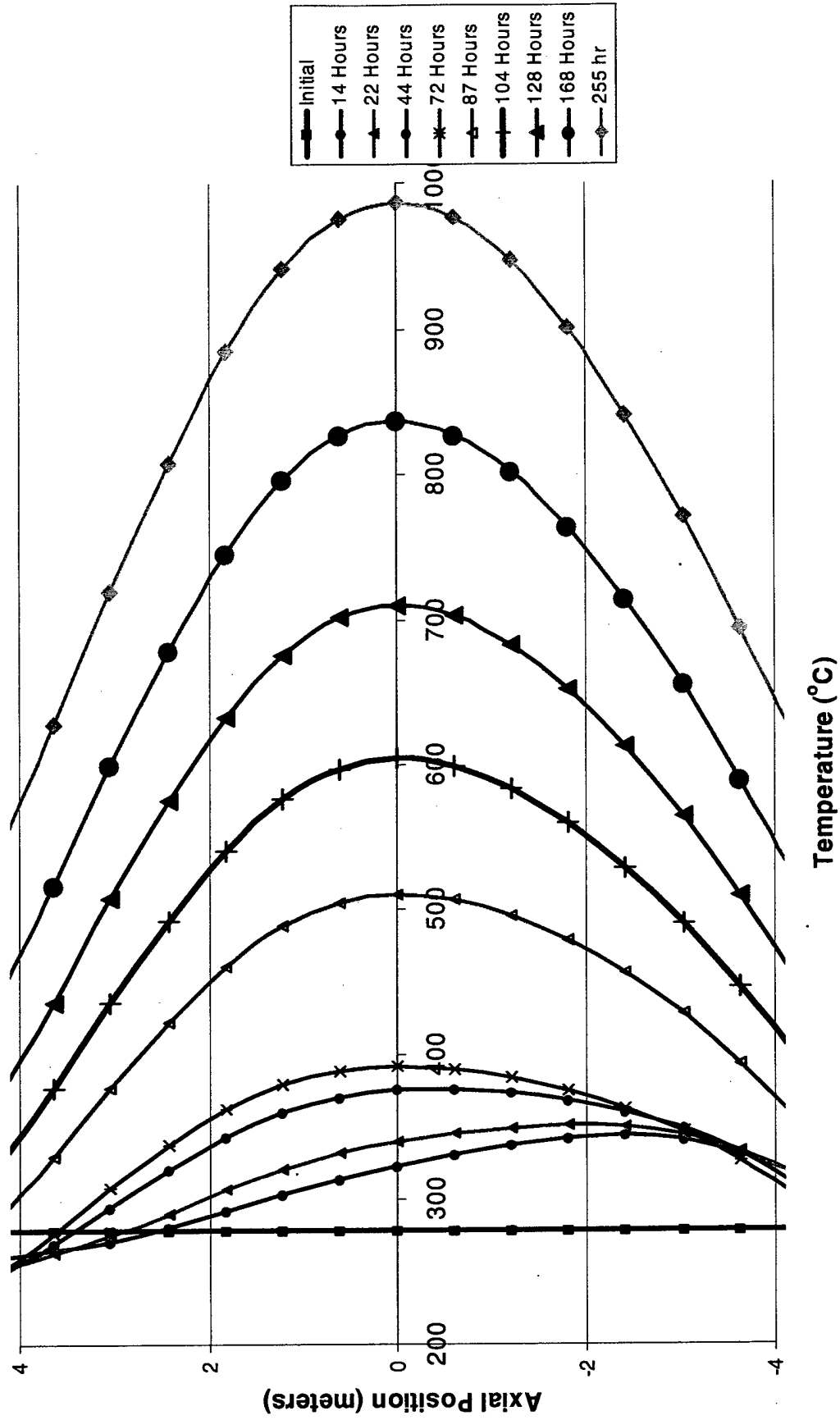


Figure 6.15 Vessel Temperature with RCCS Failing at 72 Hours

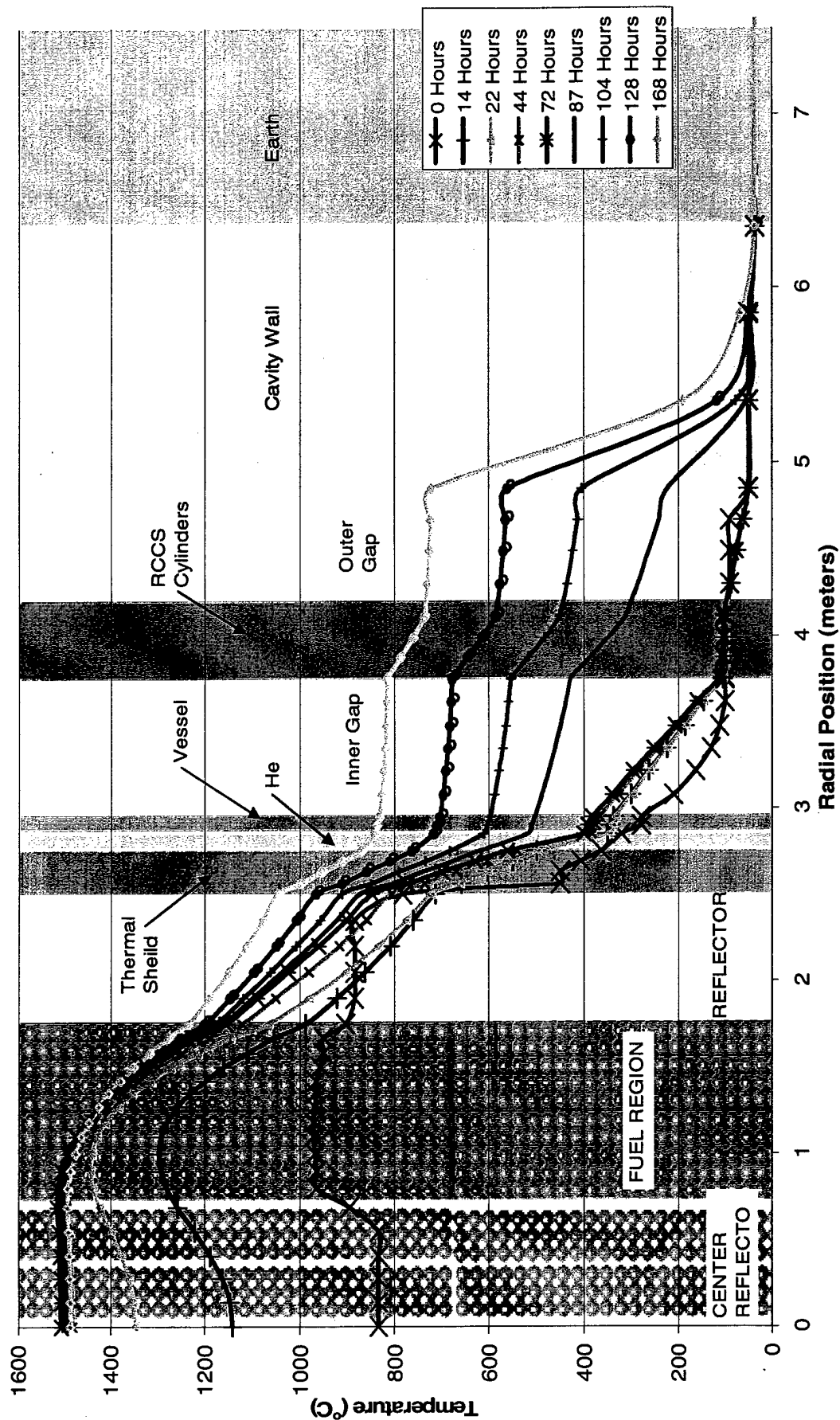


Figure 6.16 Reactor Cavity Radial Temperature Profile with RCCS Failing at 72 Hours

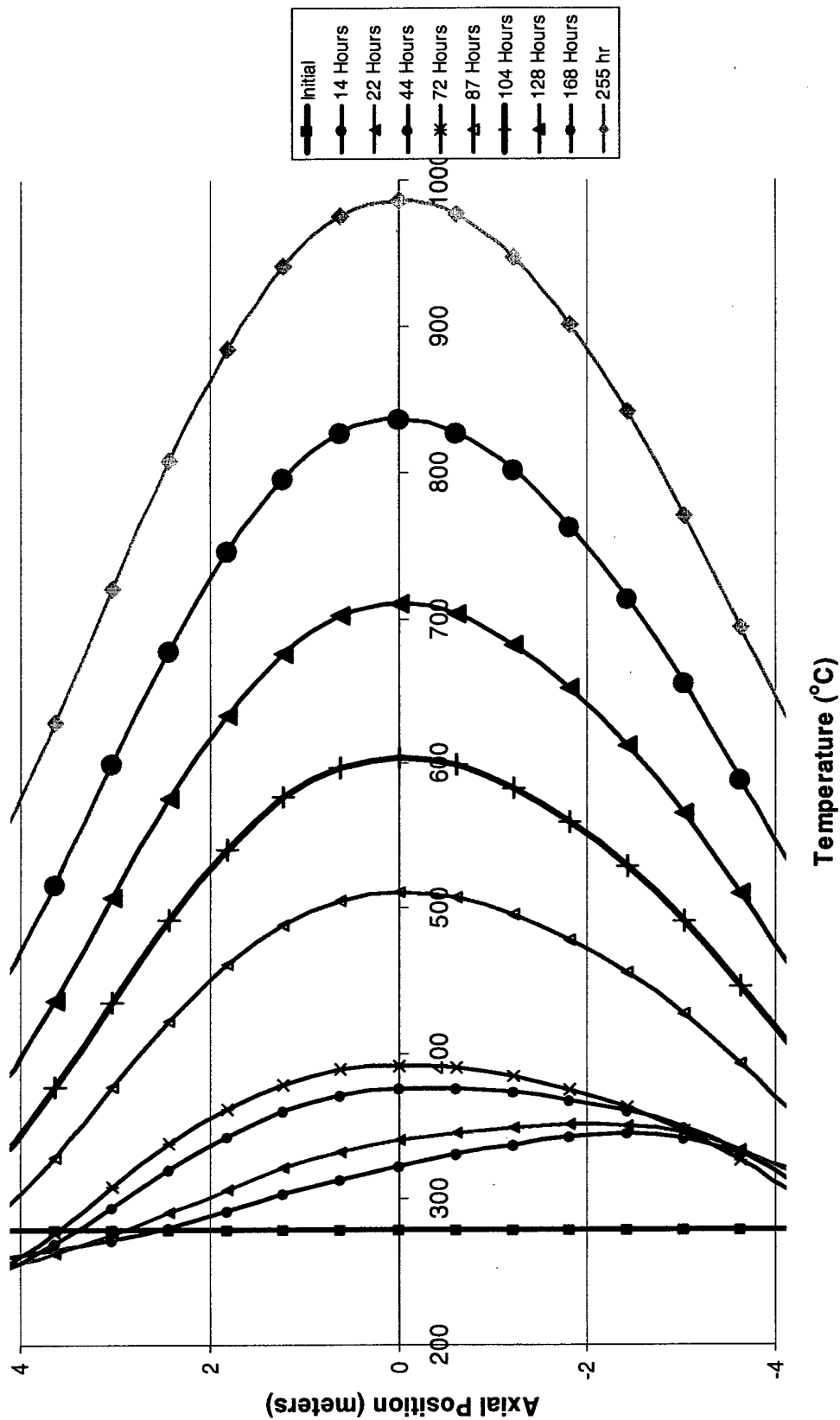


Figure 6.17 Vessel Temperature with RCCS Failing at 87 Hours

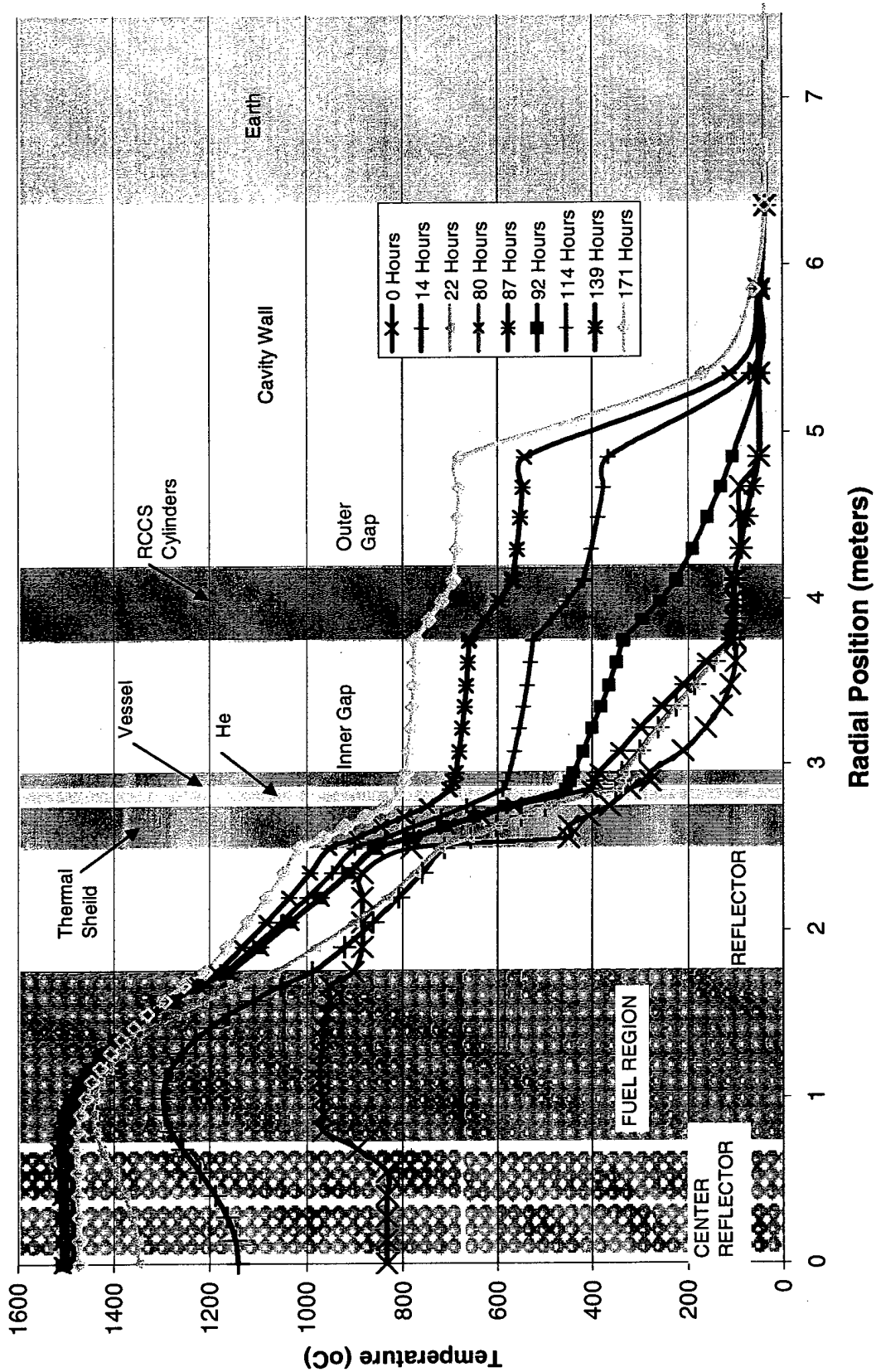


Figure 6.18 Reactor Cavity Radial Temperature Profile with RCCS Failing at 87 Hours

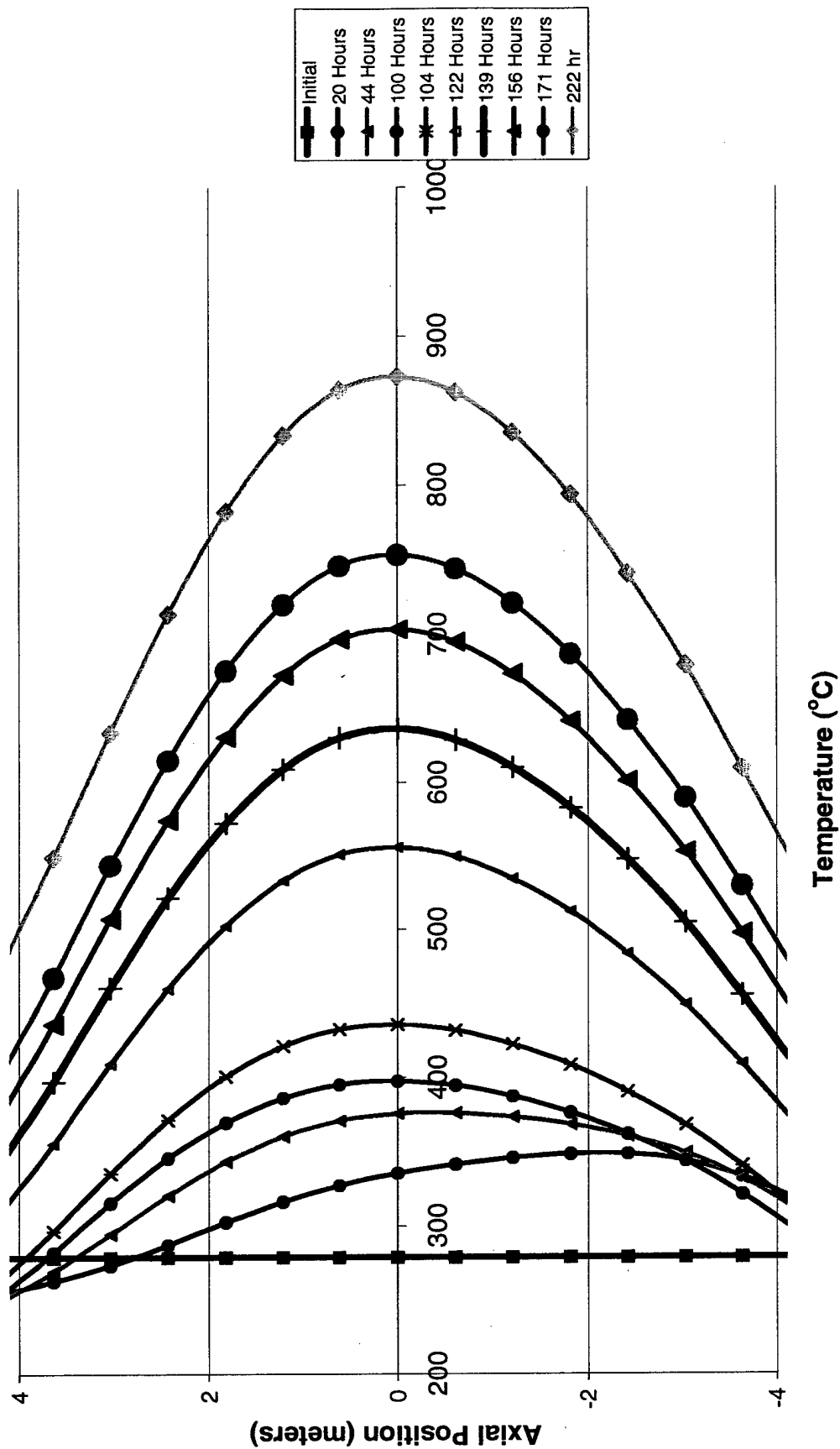


Figure 6.19 Vessel Temperature with RCCS Failing at 87 Hours

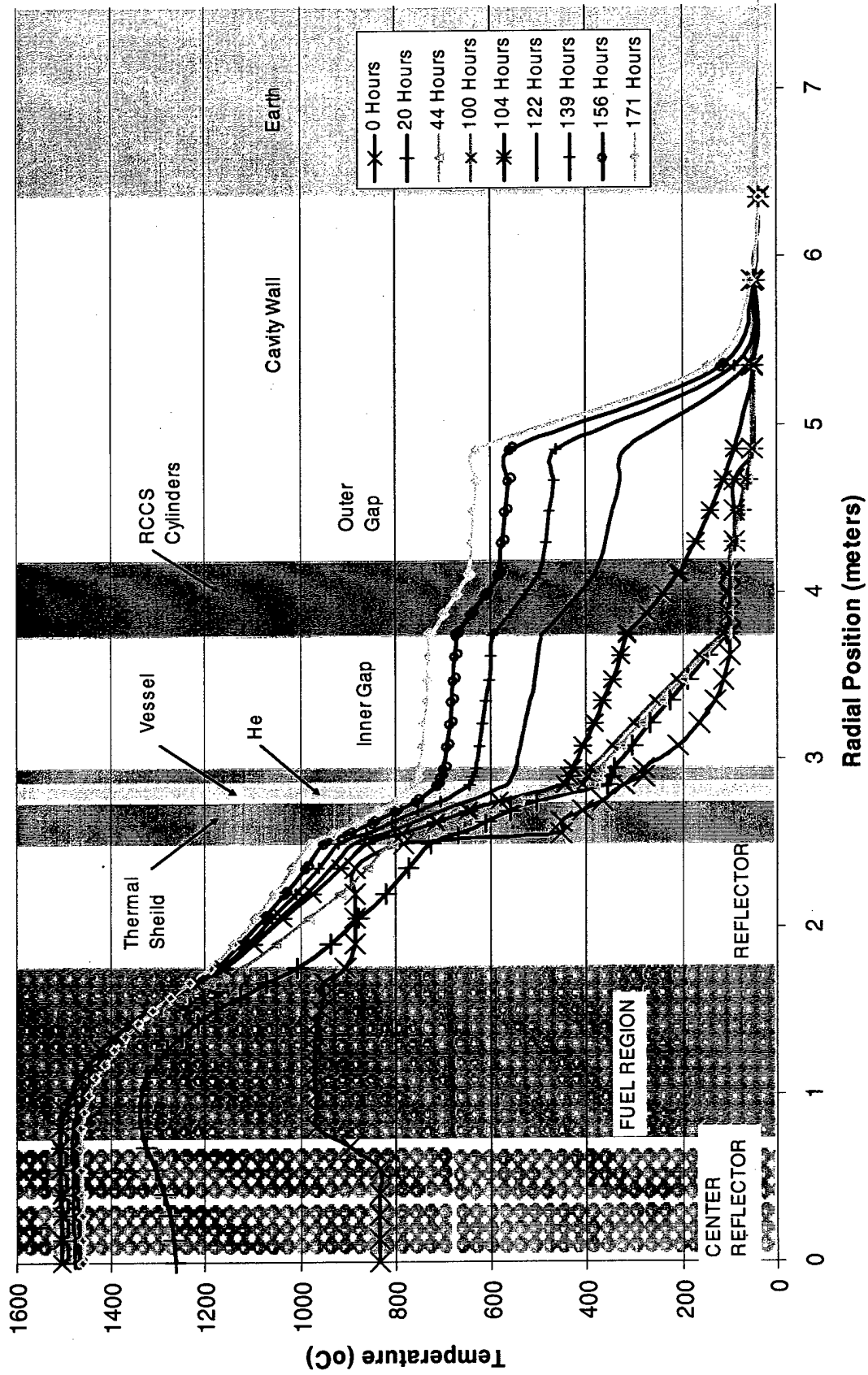


Figure 6.20 Reactor Cavity Radial Temperature Profile with RCCS Failing at 100 Hours

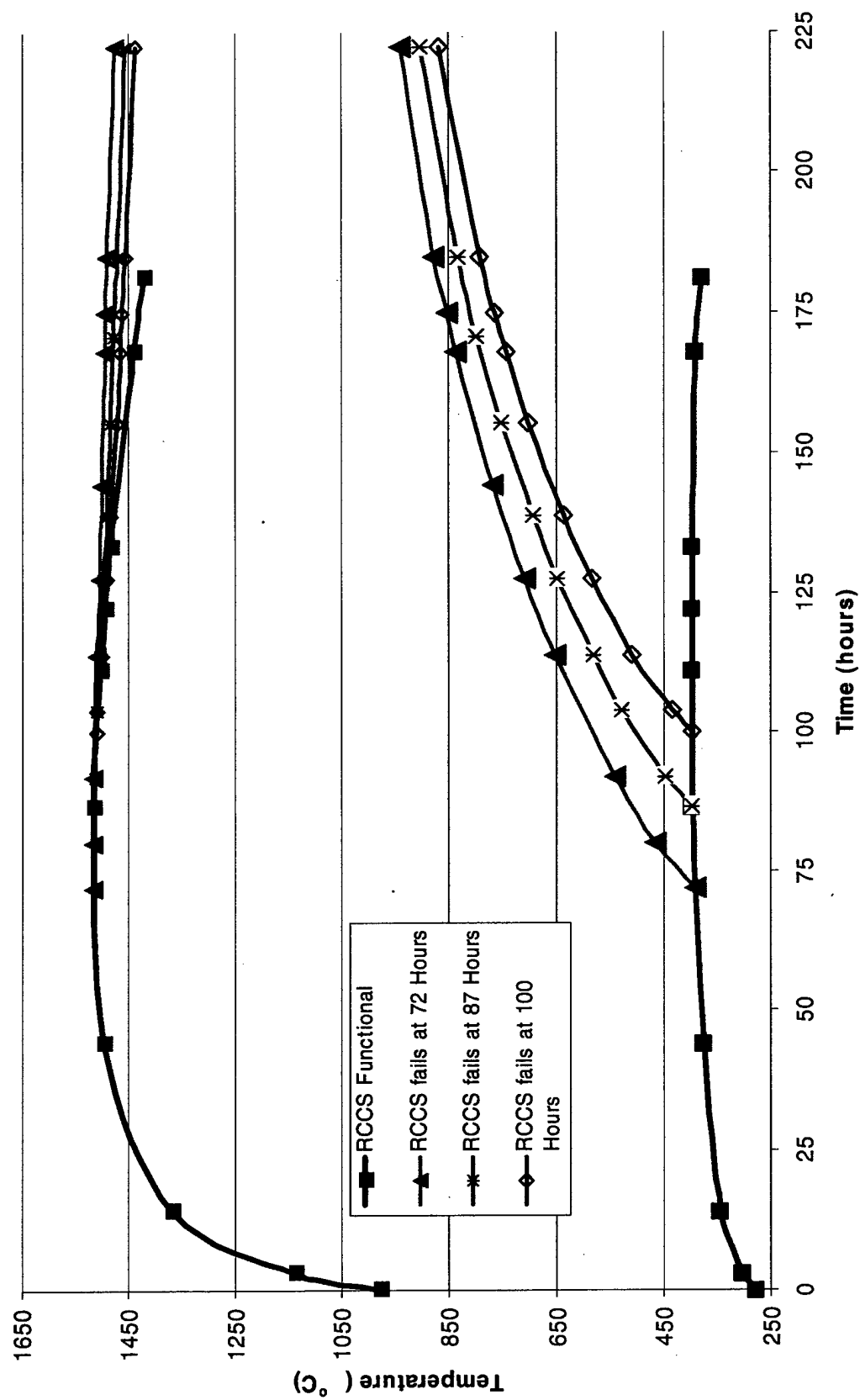


Figure 6.21 Maximum Core and Vessel Temperatures for the Four Cases

Discussion of the Thermal Analysis Results

The two previous sections dealt with the results of the thermal analysis of the PBMR for the two different casualties. Both of these casualties involve interfering with the inherent heat removal capability of the Reactor Cavity Cooling System. The steam blanketing case is shown to have little effect on the safety of the plant. The second casualty results in direct impairment of the RCCS due to operators failing to replenish the systems supply of water. The cases are not analyzed beyond the 222 hour point. In all cases with a failed RCCS the temperatures in the RCCS cylinders are high enough to cause their mechanical failure. This failure would have resulted in a change in the reactor cavity configuration; therefore, analysis beyond 222 hours is deemed not useful.

6.2 Results of PBMR Structural Analysis

Section 5.6 discusses how vessel stress is calculated. As the vessel temperature increases two things happen. The helium inside the reactor vessel undergoes thermal expansion and compresses the rest of the helium in the attached piping resulting in a pressure increase. Also, the yield strength and modulus of elasticity of the vessel material decreases as the vessel temperature increases.

Following the scram, the initial pressure in the PBMR is 4.8 MPa. This pressure increases as the temperature of the helium in the vessel rises. Correspondingly, the stress caused by the internal pressure increases. The reactor vessel will fail when this stress reaches the yield point of the vessel material. Figure 6.22 is a graph that shows the decreasing yield strength plotted against temperature. Plastic deformation begins when the yield strength reaches the vessel stress. The core temperature in the analysis was relatively independent of the vessel temperature. The majority of the pressure increase in the plant occurs as the core temperature rises to its maximum value. This increase in pressure happens before the vessel temperature starts to rise in the casualties evaluated. The estimated maximum pressure in the vessel is 6.5 MPa. This pressure results in a maximum stress of 115 MPa. The pressure stress and the SA 508 yield stress intersect at approximately 500 °C.

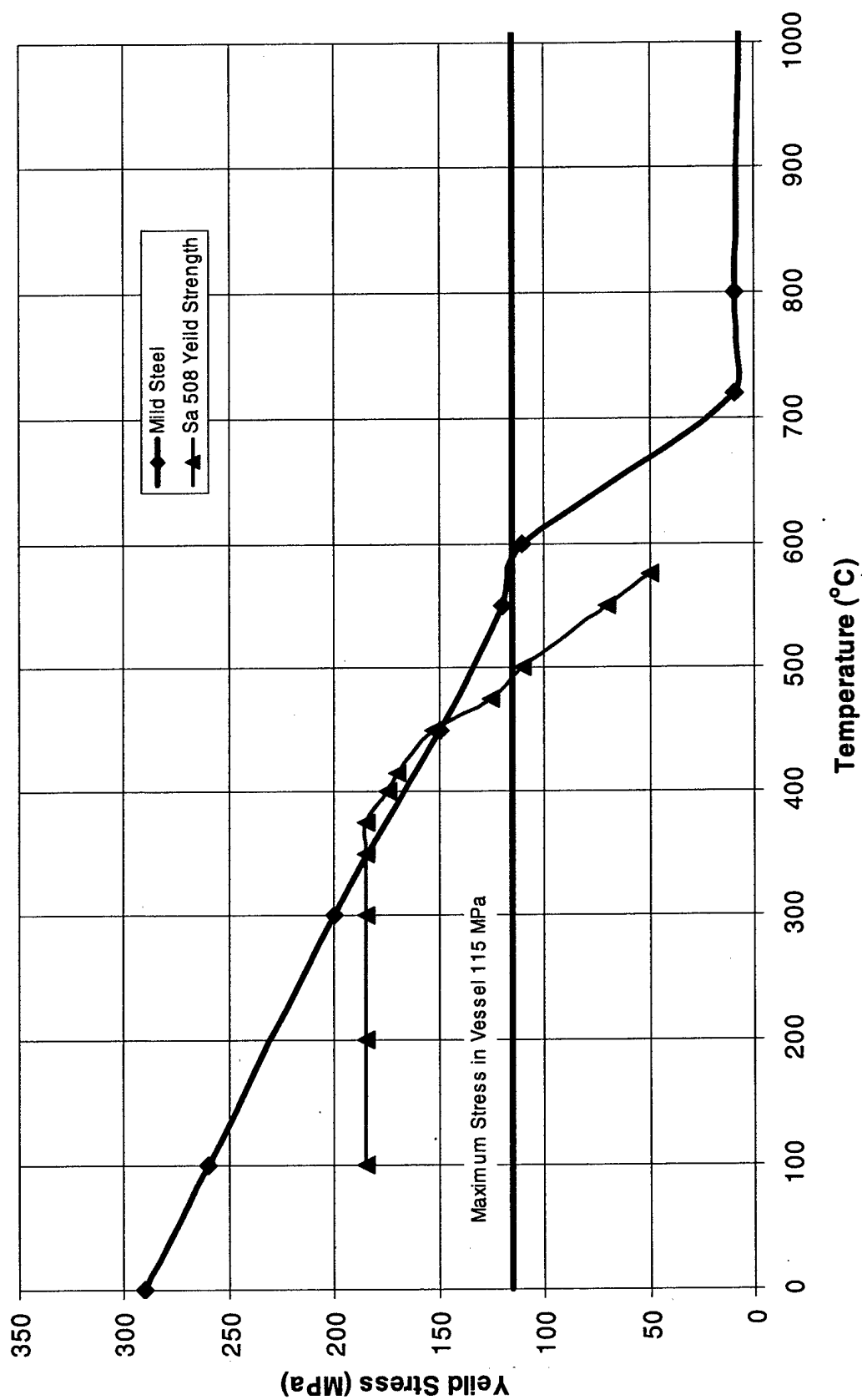


Figure 6.22 Yield Stress versus Temperature of Vessel Material

The ductile failure of the vessel will eventually lead to a rupture in the pressure boundary. As described in the previous chapter, the possibility of buckling was also examined. Figure 6.23 is a graph of the critical and ultimate stresses which will result in buckling and the maximum compressive stress due to the vessel weight. The maximum compressive stress is less than the ultimate stress up to well above 1,000 °C. Therefore, buckling is deemed not to occur in the PBMR.

When the RCCS system is allowed to expend its water supply without additional water being added, the vessel temperatures are shown to rise to above 900 °C. The strain rates of vessel material also increase with the vessel temperature. Fracture will result from the increased vessel temperature. This fracture will release the helium and reduce the vessel internal pressure. The vessel stress will become compressive. Ductile failure will likely continue near the point of fracture and the vessel will possibly collapse since the fracture will act as a stress concentrator.

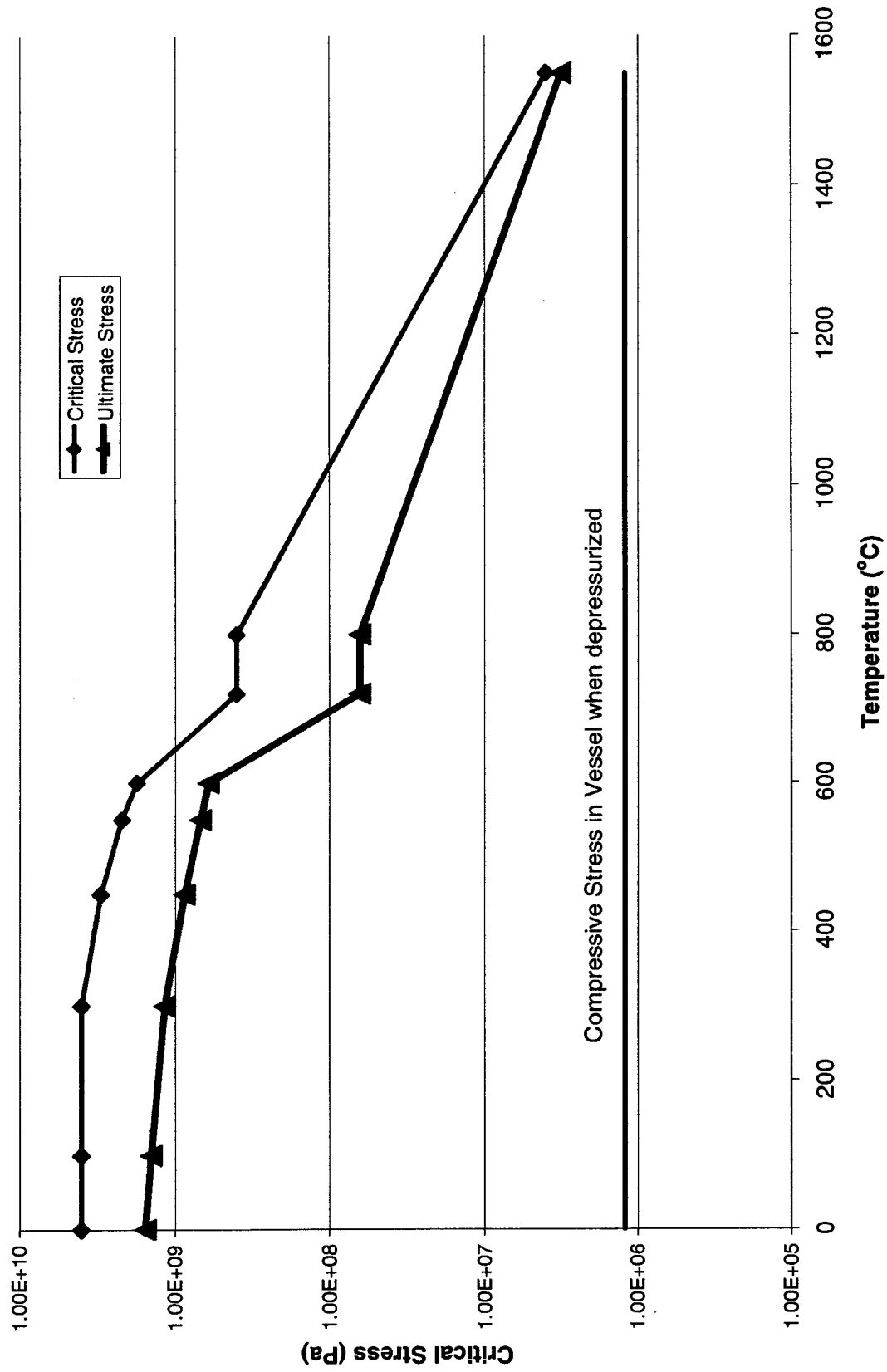


Figure 6.23 Bucking Stress in PBMR Vessel

Chapter 7 Probabilistic Assessment of PBMR Casualties

The two casualties analyzed in Chapters 5 and 6 yield very different results. The steam blanketing casualty is shown not to be of concern for the safety of the PBMR. Inaction by operators to refill the RCCS is shown to have detrimental effects on the Reactor Unit. Temperatures of the vessel are high enough to possibly cause catastrophic failure of the pressure vessel. This casualty can therefore result in the release of fission products due to damage of the fuel pebbles. This chapter deals with assigning probabilities to these casualties so that an overall judgment can be made. In an official risk-based analysis these probabilities would be assigned by establishing a commission to discuss each event in the fault tree. For this work, the author assigned the event probabilities after consultation with members of the Nuclear Engineering Department of the Massachusetts Institute of Technology.

7.1 Steam Blanketing

The addition of steam to the reactor cavity does not significantly affect the operation of the PBMR decay heat removal system. The cavity being completely filled with steam results in a 20 percent reduction in graybody radiation heat transfer. This reduction in radiation heat transfer leads to a 32 °C temperature rise on the vessel. This temperature rise is not enough to cause damage to the reactor vessel. Therefore, the probability that this casualty will result in fission product release is assigned a value of zero.

7.2 Probability of the RCCS failing due to Operator Inaction

The Reactor Cavity Cooling System can operate for approximately three days without being filled with water if it's in a no flow condition. A large earthquake could knock out the electrical power to the primary and backup pumps for the RCCS. The probability of this occurrence is associated with the likelihood of an earthquake of sufficient size occurring in the vicinity of the power plant. The probability of this occurrence is 1E-4/year.

For this casualty to have an effect on the PBMR safety; operators must not add water to the RCCS for at least 100 hours. The first variable in this casualty is the time required for operators to take the correct actions. The operators could fail to take the appropriate actions for one of two reasons. First, the earthquake could have rendered access to the PBMR site or RCCS piping impossible. Second, the operators could take improper actions relating to the RCCS. The longer it takes for operators to respond correctly, the higher the possibility of damage to the PBMR vessel. Another variable in the analysis is the time at which the RCCS ceases to function with enough capacity to remove the decay heat generated in the core. Chapter 6 details the results of the three different initiation times for complete failure of the RCCS. In reality the failure would be more gradual, but Figure 6.22 shows that the results are nearly the same for the three different cases just shifted in time.

Due to analysis of how much energy is required to boil the water in the RCCS the probability that the system fails before 72 hours is deemed to be very low. After 72 hours the probability the system failing would increase due to the fact that there is a fixed amount of water in the cooling cylinders. The mean value for the time that the RCCS ceases to function is assigned a value of 87 hours. The initial rupture of the system should occur before the vessel reaches 700 °C since at this temperature the pressure stress is 10 times higher than the yield stress. Using Figures 6.21 and 6.22 a mean time for the catastrophic failure of the vessel is estimated to be 185 hours, which corresponds to an 850 °C vessel temperature. In order to prevent the catastrophic failure of the vessel the operators must respond before this time. Therefore the probability of this event is based on the likelihood that operators will not respond properly in 185 hours. It should be noted that this time estimate could be greatly improved by more accurate mechanical property data for the vessel material and testing of stress concentrations in a ruptured cylindrical shell. The probability of the operators not acting in time is $1\text{E-}3$.

The next event that needs to occur is that the fuel pebbles must be damaged in order for fission products to be released. Several factors affect the possibility of this event. Following the failure of the vessel the fuel could be damaged by mechanical shock, thermal shock, or chemical attack. All of these events are likely if there is a catastrophic

failure of the reactor vessel. The probability of fuel damage is therefore given a probability of 0.75 if the vessel has failed.

The final event that must occur is that the fission products are transferred out of the PBMR containment area. The fission products could be entrained with helium in the cavity and carried out very the pressure relief system. The relief system could have failed open during the vent off of the helium. The rupture disc in the vent pipe could have blown and its protective valve failed open. Any of these occurrences could result in fission product release to the environment. The probability of this event occurring assuming that a large earthquake has already occurred is 0.001.

The probability of fission product release is obtained by combining all the probabilities above. The calculated probability for this casualty affecting the surroundings is therefore $7.5E-11$ /year. This probability would be combined with all the other possible casualties that could lead to fission product release to the environment. These other casualties would have to be analyzed and assigned the appropriate probabilities of occurrence.

Chapter 8 Conclusions and Future Work

The purpose of this work was to analyze the PBMR reactor to determine if its inherently safe design warranted use of a non-standard containment system. This determination was to be based on the probability of fission product release for casualties which impaired the decay heat removal capability of the reactor. The initiating event for both of the casualties was a beyond design basis earthquake.

The first casualty was steam blanketing of the Reactor Unit by water released from a ruptured RCCS cylinder. The deterministic analysis portion of this work showed that the reduction in radiation heat transfer due to the presence of water vapor was negligible, and had no effect on reactor safety. The second casualty was the loss of the RCCS due to operators not replenishing the water. This event resulted in vessel temperatures in excess of 900 °C. If proper action was not taken within a given amount of time catastrophic failure of the reactor vessel could occur. The likelihood of this event was assessed to be $7.5E-11$ /year. This risk would be combined with the risk of all the other possible occurrences that could result in fission product release to obtain the total risk to the public. The analysis in this work has shown that the Eskom assertion of the PBMR not requiring a standard containment system is justified for the evaluated casualties.

Here are some recommendations for further analysis into risk based licensing of the PBMR. Future efforts should be directed into further identifying casualties that are specific to PBMR reactor design. A Computational Fluid Dynamics program should be used in follow-on analysis to overcome the limitations of HEATING. A group of experts should be involved in assigning the probabilities to the individual components of the casualty risk analysis.

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Appendix A: RCCS Addition Information [5]

The purpose of the RCCS is to transfer to the sea (or to the atmosphere, when the sea water system is not available), the heat from the reactor cavity during all modes of reactor operation, including a Pressurized or Depressurized Loss of Forced Cooling (PLOFC or DLOFC) event, thereby protecting the concrete wall of the cavity, and ensuring that the maximum Reactor Pressure Vessel (RPV) and fuel temperatures are not exceeded.

Natural processes, for instance, thermal radiation and convection, transport the heat from the reactor vessel walls to the cooling chambers of the cavity cooling system.

The RCCS is a three-loop system of large water chambers surrounding the reactor. Heat absorbed by the water is pumped to plate heat exchangers where it is transferred to the UHSS. The UFCS is a water-cooled system operating in parallel with the RCCS, and using the same pumps and heat exchangers as the RCCS. Water can be directed to either the RCCS or the UFCS, or to both, and both are backed-up by the cooling towers. Both can operate in passive mode when required.

The UFCS consists of an annular tank with a water jacket. Decay heat from the fuel is transferred to the water in the jacket, the hot water then being pumped to the RCCS heat exchangers.

The RCCS is designed to remove the specified waste heat from the reactor cavity under all operating conditions. It is a water-based system made up of proven plate heat exchangers, pumps, valves and pipework. All operating pumps have standby units which will automatically come on line in the event of a failure of an operational unit. Valves are generally kept in the open position, being closed manually for maintenance purposes.

In addition to having back-up cooling towers, the RCCS has a passive mode of operation in that water contained in these systems can be allowed to boil off over a period of approximately three days, sufficient to allow the fuel heat to decay to a safe level. Additional water from an outside source can also be introduced to replenish water lost through evaporation. The period of three days is based on 50% of the water boiling away, the heat load being 1 100 kW as expected under DLOFC conditions.

The systems are fitted with temperature, pressure and level indicators, and facilities are available for extracting samples on a regular basis.

All components of the RCCS and UFCS have a design pressure rating of 1 MPa, while operating pressures are not expected to exceed 0.5 MPa.

The schematic layout of the RCCS and UFCS is shown in the figure on the next page. The RCCS consists of three identical cooling trains, where each train is an independent, low-pressure, closed-loop, pump-driven, water-based cooling system

Within the reactor cavity, the system consists of a series of 45 low carbon steel water chambers, each approximately 500 mm in diameter and 22 m long, arranged vertically around, and concentric to, the RPV. The chamber wall thickness is 10 mm. Three 150 mm diameter inlet and three outlet headers are arranged around the top of the chambers. Each outlet header transports hot water to a heat exchanger situated outside the reactor cavity, after which it is pumped back into the corresponding inlet header. From each inlet header, feed lines enter the top of every third chamber, then run down the inside of the chamber to discharge cold water at the bottom.

The entire chamber structure is suspended from a support ring resting on the concrete wall of the reactor cavity, thereby allowing free expansion and contraction. Anti-syphoning devices are fitted to prevent the chambers from being emptied in the event of a low-level break in the pipework outside of the reactor cavity, and the trains are connected to fixed water make-up lines fed from the existing Koeberg facilities.

Each cooling train has its own plate heat exchanger and two 100% pumps. These are situated on the -19 m level of the Module Building. The three trains are backed up by two 50% cooling towers situated on the roof of the module. Each tower has two 100% main circulating pumps and two 100% spray pumps.